NUCLEONICS

May 1953

New policies on industrial atomic power are being studied by Congress with advice from AEC. Leaders in this planning are Rep. W. Sterling Cole (left), chairman, Joint Congressional Committee on Atomic Energy, and Gordon Dean (right), chairman, AEC (see p. 71)

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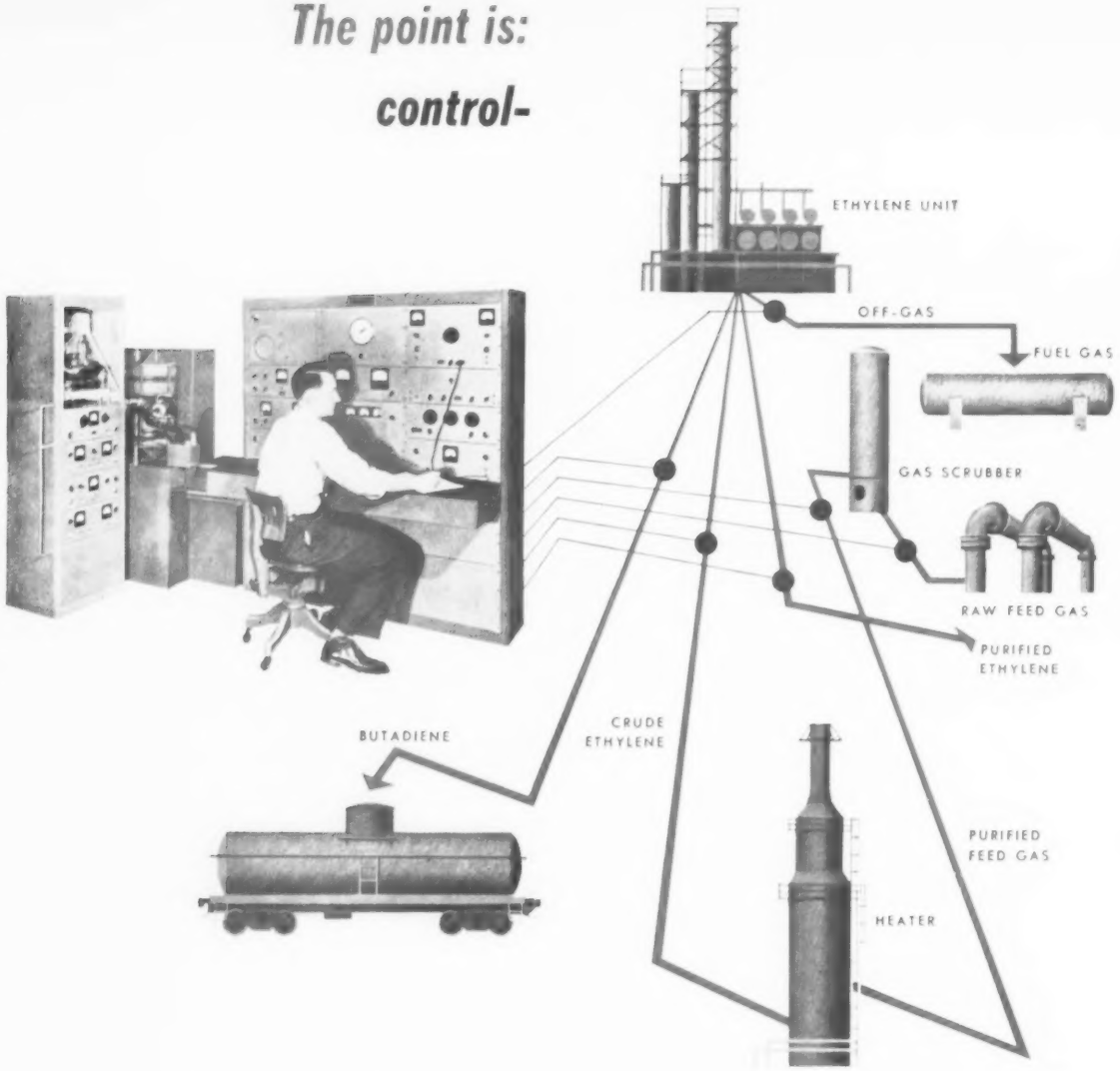

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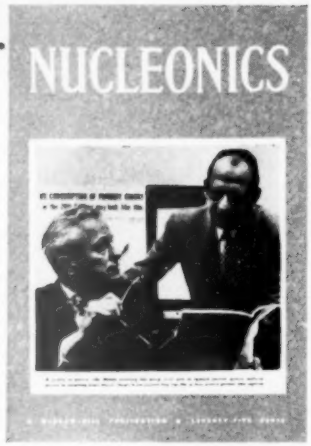

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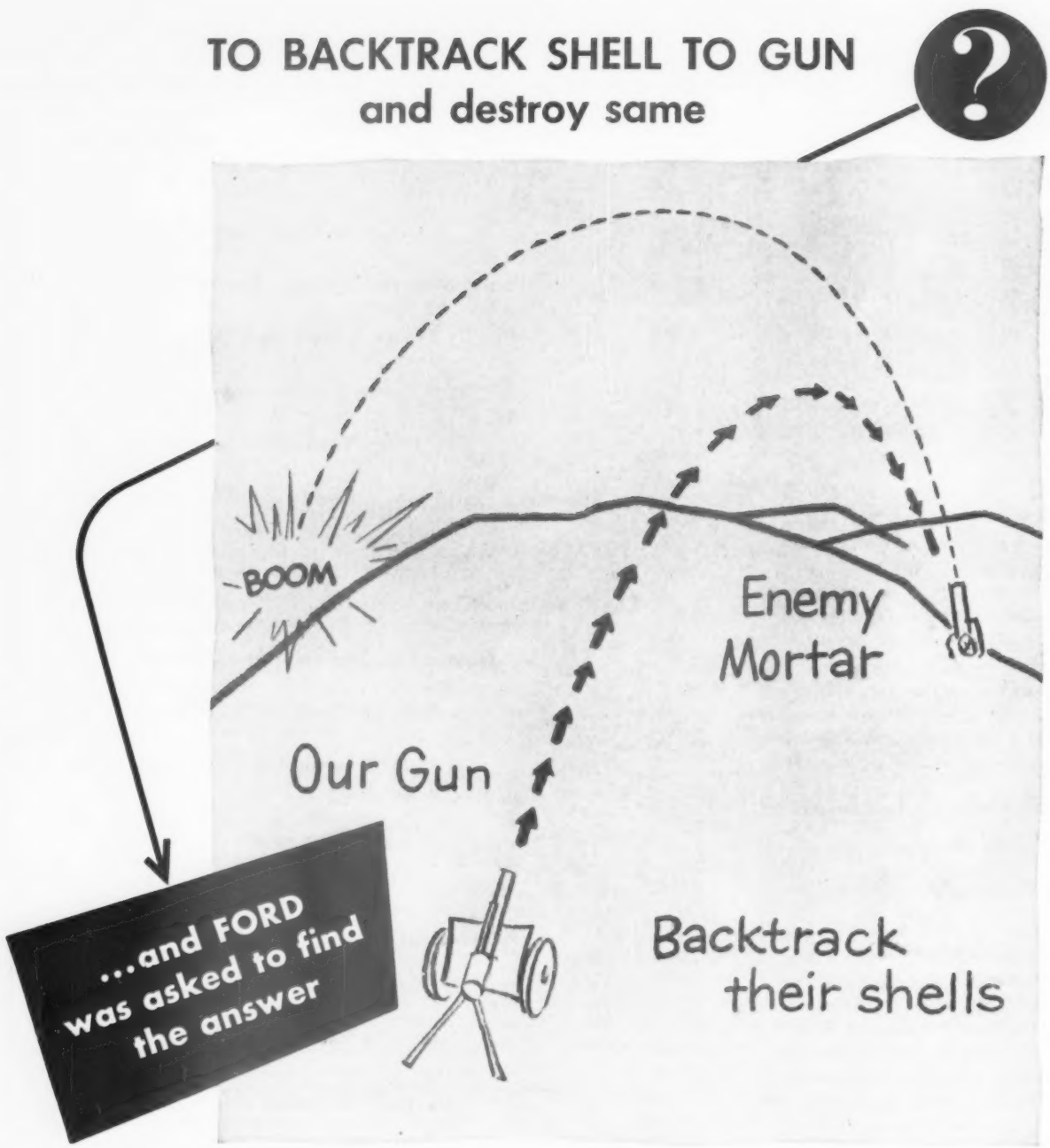

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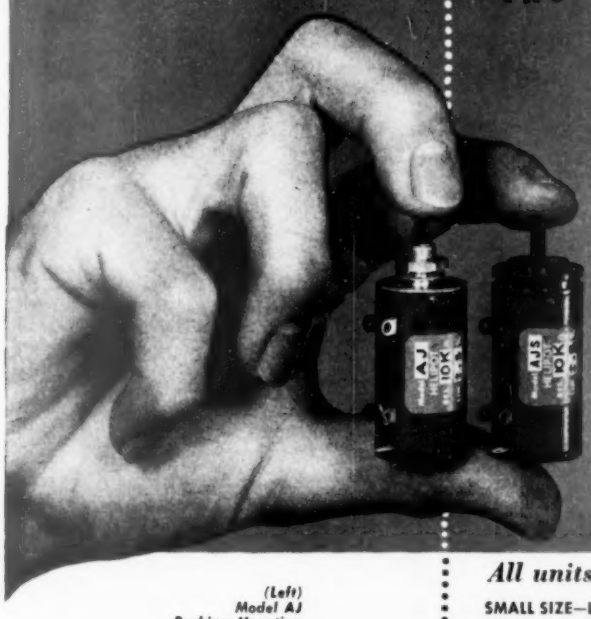

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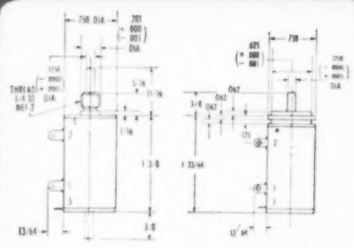

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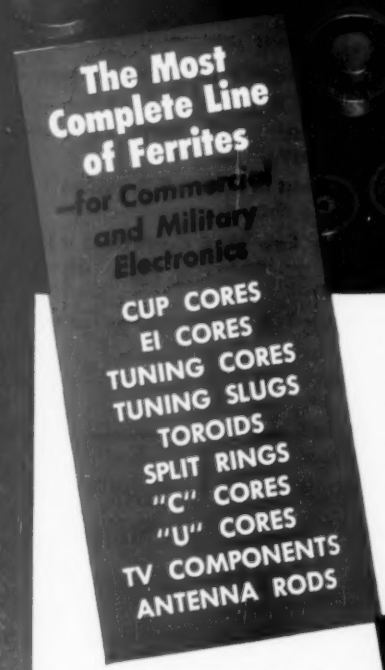

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*Max. Perm.	–	100	183	1100	1030	1710	3300	4300	3800	3000	750
*Sat. Flux Density	Gauss	1500	1900	4700	3100	3800	3200	3400	2800	2000	2900
*Residual Mag.	Gauss	1000	830	2700	1320	1950	1050	1470	1500	700	1600
*Coercive force	Oersted	5.0	3.0	2.1	1.0	.65	.25	.18	.35	.30	.80
Temp. Coef. of initial perm.	%/°C	.15	.04	40	30	25	1.3	.66	.80	.30	.22
Curie Point	/°C	300	260	330	165	160	160	150	125	70	180
Vol. Resistivity	ohm-cm.	$1x10^7$	$2x10^5$	$2x10^4$	$3x10^7$	$4x10^5$	$1.5x10^6$	$1x10^4$	$2x10^4$	$2x10^5$	$5x10^7$
Loss Factor: At 1 mcs/sec	–	.0005	.00016	.00007	.00005	.00008	.00008	.00030	.0004	.0003	.000055
At 5 mcs/sec	–	.0007	.0011	.0008	.0012	.002	.0007	.00155	.001	.005	.0004

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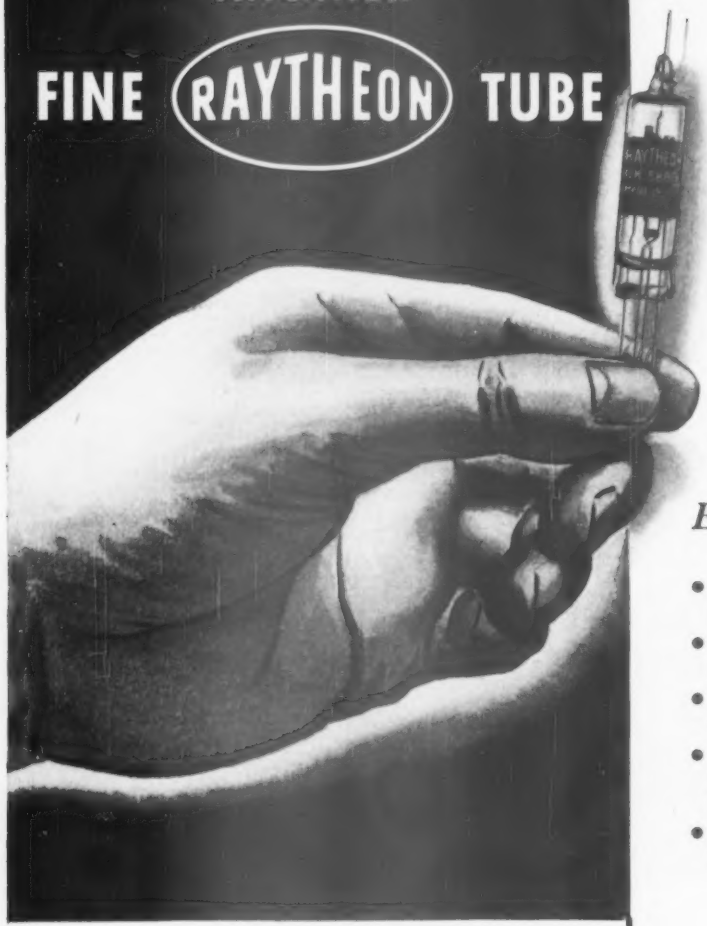

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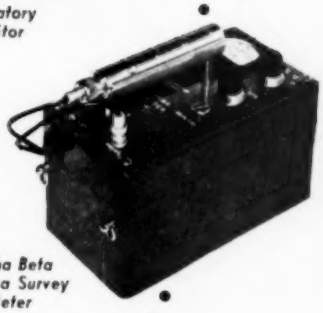

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NUCLEONICS MAY, 1953

Jerome D. Luntz, Editor

Atomic Forum

THERE HAS BEEN a lot of talk in recent months about time scales for the full-fledged realization of the peacetime potential of atomic energy. Most of this talk has hinged on when we will have economically competitive nuclear power.

Although the date of this event will be a much celebrated one—whether it be 5 years or 10 years from now—atomic energy is already affecting a large segment of American industry in a substantial way. In fact, although industry is only just now beginning to invest its own money in this field, we are already well along in the development of an atomic industry in its own right.

If evidence of the existence of this industry is needed, it came last month with the formation of this country's first association of companies interested in the advancement of atomic energy —the Atomic Industrial Forum, Inc. (see p. 70).

Organized by a group of leading executives connected with groups active in the atomic energy program, the purpose of the Forum, according to its directors, is as follows: It will serve as a meeting ground for all those interested in the practical aspects of atomic energy. Its principal function will be the accumulation, exchange and dissemination of information. It will be the voice of industry in this field. But, although major revisions in atomic energy legislation are now being considered, the Forum will not lobby, will not as a group seek changes in the law.

THIS ASSOCIATION has been set up at a most appropriate time. The need for something like it has been growing in recent months—a need for a unified group to serve very diverse technical, economic, and legal interests.

Even if the Forum doesn't make recommendations on legislation, it can serve a very valuable function as a catalyzer of vigorous discussions of important legislative issues, e.g., methods of safe and secure industrial use of nuclear fuel, the question of a "free market" for nuclear fuel, safety problems in plant location, patents, etc.

In addition, we hope that the Forum will organize at least one technical conference a year. For some time, we have felt that there is a need (see "An Annual Nucleonics Meeting," NU, p. 2, June '51) for a live forum for the cross-fertilization of ideas coming from people of quite varying backgrounds working in this field.

Other functions which the Forum might serve could include a public relations program to engender public confidence in the safety of atomic energy, particularly with respect to nuclear reactors and uses of radioactivity. In this regard, the Forum might set up a "safety and standards" group to make recommendations on all aspects of the use of atomic energy.

ONE ASPECT of the Forum's operations which must be clarified is the proposal (subject to approval at first meeting of directors on May 11) that voting members be those who have been previously cleared for access to classified information. It is said that this was done to assure that the leaders of the group would be actively involved in atomic energy work and that it is not intended for any activities of the group to be in secret.

We feel very strongly that, although there is no doubt about the desirability of voting members being active in the atomic energy field, there should be no requirement of clearance. There are many working in this field who do not need clearance for their work, e.g., those involved in uranium mining and fission product utilization. These people should not be deprived of playing a leading role in this group. In addition, AEC would effectively act as the "admissions committee" for the Forum because AEC passes on who gets clearance and who doesn't.

If this association is to make a significant contribution to the industry, its appeal must be broad-based. It must be committed to the principle that this industry can grow only if it can have information and plenty of it. All companies must have an equal crack at atomic energy. One never knows what the source of significant contributions to the growth of the industry will be.

The association can play an important role in the fostering of this growth. NUCLEONICS hails this pioneering group and wishes it well.

The Key to Atomic Power

A low-cost reactor can be achieved by inventive design with relatively minor advances in present technology. Uranium will be abundant and cheap enough to encourage the nuclear power industry

Industrial Nuclear Power

Because of the current interest in deriving useful power from atomic energy, NUCLEONICS has gathered together the four articles presented in the next eleven pages. Included are a consideration of the best route to useful power, a survey of foreign markets for package power (p. 14), recommendations for changes in the Atomic Energy Act (p. 16), and a review of recent advances in reactor technology (p. 18).

By KARL COHEN
Vice President
Walter Kidde Nuclear Laboratories, Inc.
Garden City, New York

THE PROPER ROLE of private initiative in atomic development is now being seriously reconsidered for the first time since the Congressional hearings of 1945–1946, which preceded the passage of the McMahon Act. Public and congressional support for broader industrial participation in atomic development is principally sustained by the prospect of civilian atomic power, since economic atomic power is not unnaturally associated with commercial plant operation. For this reason, studies of the economics of atomic power have unusual present interest.

Most observers conclude that power cannot be produced economically without major advances on present technology, except as a by-product of plutonium manufacture. This indicates, to one school of thought, the necessity for a program on breeder reactors; to another, the need for a guaranteed price of plutonium; and to still others, government financing of the reactor. These diverse points of view lead to equally diverse opinions on the urgency and method of improving the McMahon Act; an extreme view is that a more favorable economic forecast should be a prerequisite to amending the law.

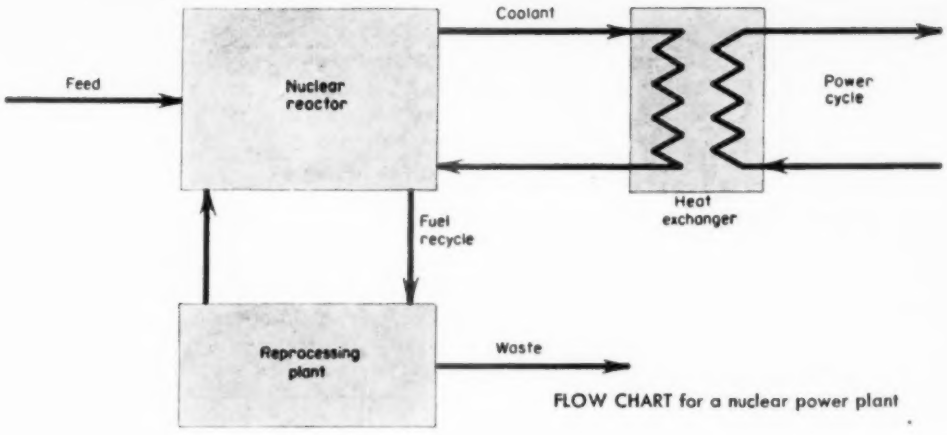
The present article takes a more optimistic view of the economic possibilities. We believe that dual-purpose reactors, although offering immediate economic feasibility, are not the only, or even the preferred method of approach. There is reason to believe that single-purpose plutonium reactors will produce cheaper plutonium and that single-purpose power reactors can produce economic power.

The substance of this article will be a careful analysis of the economic assumptions on which power cost estimates have been based. What is perhaps not fully appreciated is that the various experts base their cost estimates on different assumptions; and the conclusions depend strongly on the assumptions. This is not merely a question of disguising true and official figures for security reasons. A theme of the present paper is the slightly heretical one that official cost figures for government operations, even if known, would not always be the appropriate basis for cost estimates of commercial power.

Cost Elements

Operation of a nuclear power plant is shown schematically by the illustration on the next page. Feed, which may be fissionable or fertile material, or a mixture of the two, is fed into the reactor at a rate sufficient to maintain the reactivity. Spent fuel is recycled through a chemical and/or metallurgical reprocessing plant at a rate determined by one of a number of factors: radiation damage, loss of reactivity, etc.

Waste material is drawn off from the reprocessing step. The waste material is likewise a mixture of fertile and fissionable materials and may, in some instances, be more valuable than the feed; the use of the terminology "waste" and "feed" indicates merely the direction in which these products flow. For completeness an isotope separation step might be included in



FLOW CHART for a nuclear power plant

the reprocessing cycle, but this can be omitted without real loss of generality.

The reactor also produces heat, and this is transferred through a heat exchanger to the working fluid of a power cycle. It is approximately correct to say that the power cycle, from the heat exchanger on, would be the same as the power cycle of a coal-fired plant. There is some indication that the economic operating pressure for steam plants with low fuel costs is less than that of coal-fired plants, but for the purposes of concentrating attention on the reactor, it is expedient to ignore this difference.

The illustration applies equally to the steady-state operating condition of reactors which burn U^{235} and convert fertile material to fissionable material, and to breeder reactors which would consume fertile materials only and produce fissionable materials. The relation between recycling and purging to waste is variable; the once-through reactor is represented in this scheme by the limiting case of zero recycle.

The contribution to power costs of the plant components on this flow sheet are given in Table 1, which is discussed in detail under "Single-Purpose Reactors." In Table 2 the corresponding costs for the parts of a conventional power plant which the reactor and heat exchanger would replace are listed.

When we try to put numbers into Table 1, we run into difficulties, because there is no general agreement about the costs of the reactor or the reprocessing steps which would be appropriate for private operations. To surmount the present difficulty of a complete lack of commercial experience, the plutonium-plus-power reactor proposal has been advanced. In this way it is hoped to sidestep the economic problem.

Dual-Purpose Reactor

The central issue here is what value to assign to plutonium. The proponents of a dual-purpose liquid-metal-cooled reactor write (*1*):

" . . . preliminary estimates of the cost of plutonium production in a dual-purpose reactor of this kind were made and compared with the cost in present-day reactors designed to produce plutonium only.

"Quite without regard to power production, it turns out that the liquid-metal cooling represents, in itself, an advance in technology which promises a reduction in costs.

"Because of the high cost of plutonium, the reactor should be designed and operated so as to produce it as cheaply as possible regardless of the effect on the efficiency of the power production. It turns out, however, that one of the big advantages of liquid-metal cooling is that, after plutonium production has been optimized, it is still practical to generate power. It is in this sense that the two products are economically compatible.

"Unless some still cheaper method is found first, present-day government plants would serve as the basis for determining the competitive price of plutonium. In comparing government costs with private industry, the same bookkeeping methods would have to be used along with the same tax computations and depreciation periods."

This answer not only sets a price on plutonium, but also makes the costs of uranium and chemical reprocessing immaterial, since they are part of the costs of the yard-stick plants. From the standpoint of the government, the principal unsatisfactory feature of the solution would appear to be a requirement to guarantee the market for plutonium (*2*). Such an objection would, of course, disappear with an upward revision of the estimated plutonium requirements.

There are other difficulties which appear to be more fundamental. One is suggested in the phrase "unless some still cheaper method is found first." Here we have room for endless speculation on the relative (paper) merits of various reactors.

Another difficulty is revealed by the phrase "the big advantages of liquid-metal cooling is that, after plutonium production has been optimized, it is still practical to generate power." This means that with some coolants, optimum plutonium production is not compatible with power generation. Taking the two phrases together we have an indication that lowest price plutonium may not be associated with power generation at all—a proposition that certainly takes the bloom off dual-purpose reactors.

Single-Purpose Reactors

It would seem from the preceding discussion that the dual-purpose reactor concept may not automatically eliminate all present problems. Let us, therefore, consider single-purpose reactors for the production of power and see how far away we are from economic feasibility. (Refer again to Table 1.)

The obvious proposition for nuclear fuel is to strive to make the fuel cost negligible and then to use up the available credit as capital cost of the nuclear

TABLE 1—Costs of Nuclear Power*

Investment	Case A	Case B
Reactor and heat exchanger		
Reprocessing plant		
Fuel inventory		
Total	$225/kw	$210/kw
Operating Cost (mils/kwh)		
Reactor and heat exchanger operation	0.3	0.3
Reprocessing plant operation	0.3	0.3
Capital charges	4.5	4.2
Feed cost	—	0.3
Credit for waste	—	—
	5.1	5.1

* Excluding turbine and electrical generator. 80% service factor.

TABLE 2—Costs of Conventional Power*

Investment	
Boiler	
Fuel handling equipment	
Total	$80/kw
Operating Cost (mils/kwh)	
Boiler operation	0.6
Capital charges (at 14%)	1.6
Fuel cost (coal at $7.50/ton)	2.9
	5.1

* Excluding turbine and electrical generator. 80% service factor.

reactor. For example, if as in Case *A* all fuel charges were zero (that is, in Table 1, fuel cost less credit for waste equals zero) and if the costs of operating the reactor and chemical plant were no more than 0.6 mil/kwh, we could afford 4.5 mil/kwh of capital charges, which is equivalent to an investment of $225/kw.

The $225/kw has to be allocated to three items, and this is not easy to achieve. Values quoted in the literature vary from as low as $280/kw (*3*), to $5-600/kw (*4*), to an astronomical $1,400/kw (*5*) for the ship propulsion reactor. It would seem from these numbers that we must squeeze both fuel costs and reactor and reprocessing costs to the limit to achieve economic nuclear power. In this way, one appears to be led to consider the design of advanced breeder reactors of a novel type, whose realization is some years away.

However, other lines of attack are possible. We start with the observation that if uranium is basically a cheap fuel, it is not entirely logical to do the utmost to economize it. Suppose that instead of trying for a zero fuel charge, we accepted one which would be one-tenth that of coal, i.e., 0.3 mil/kwh. In this case, the allowable capital investment for the reactor plant would be $210/kw, (Table 1, Case *B*) which to be sure is slightly reduced but may be easier to achieve. We might, for example, be able to lower the chemical reprocessing plant cost—or in exceptionally favorable cases even eliminate it entirely; we might be able to reduce the fuel inventory or use a cheaper fuel; and we might be able to build a cheaper reactor. The most interesting possibility is that some of these things could be accomplished without waiting for a brand new technology.

Let us see how much uranium we have to consume to achieve 0.3 mil/kwh. The fission of 1 gm of uranium gives about 6,000 kwh of electricity; at 0.3 mil/kwh we can pay $1.80/gm or $800/lb of fuel used. This rules out, to begin with, the use of pure fissionable materials, whose costs have been quoted (*5*) at $20/gm. Let us suppose, following Zinn (*6*), that we use natural uranium as fuel in a regenerative reactor, that radiation damage will not cause trouble until 1% of the uranium is consumed, and that we must then withdraw and discard the irradiated fuel, thus eliminating all chemical reprocessing charges.

In this case 0.3 mil/kwh fuel cost would require uranium at $8/lb. If the uranium cost were higher, we would have to irradiate longer, or accept a higher fuel charge. What is clear from this hypothetical example is that the fuel for this type of operation will have to be wholly, or largely, natural uranium; and that the price of uranium will have a marked effect on the economics of the whole enterprise. To evaluate the potentialities of this reactor concept, we shall be obliged to take a closer look at the availability and price trend of natural uranium.

The Price of Uranium

Several recent authors (*4, 6, 7*) have been satisfied to use the figure of $35/lb for uranium metal in a reactor, but Young has quoted (*8*) a figure of $10/lb, and a recent *New York Herald-Tribune* dispatch (*9*) gives the Canadian price as $7.25/lb of U_3O_8, which would be more in agreement with the latter figure. The key to this apparent discrepancy is the observation that there is no free market and that we are talking of production costs and not prices. Uranium is currently produced from high-grade deposits (Belgian Congo, uranium content of the order of 10%), medium grade deposits (Canada, uranium contents up to 1%) low-grade deposits (Colorado, order of 0.1%), and as by-products of other mining operations (Florida phosphates, 0.01%). It is plain that the American producers have the highest costs. Does the effort to develop the high-cost areas necessarily mean that the military demand for uranium is greater than the potential production of all low-cost producers? This conclusion is not certain. The strategic advantage of having production facilities in the continental United States might warrant support of high-cost operations even if additional low-cost production could be developed overseas. Unless we postulate an indefinite expansion of military demand, new discoveries will result in availability, at least for a few decades, of low-cost uranium for industrial use.

The case of indefinite expansion of military demand would perhaps push the uranium price up; but this would be associated with a concomitant expansion of demand for plutonium. One could then expect a price for plutonium which would pay the freight. There is a logical inconsistency in simultaneously assuming a static plutonium demand *and* a high price for natural uranium for commercial use.

In the price which the government pays for uranium from low-grade ores, we have a fine example of the truism that a perfectly valid price for the

government may not be the commercial cost of a product.

The Uranium Supply

Let us begin with an analysis of the availability of uranium. There are two equally relevant facts to be considered in evaluating the energy potential of a fuel. One is the current rate of production; the second is the estimated recoverable reserve. The second figure has been estimated (*10*) at 25-million tons of uranium,* or an energy reserve in U^{235} alone of 12 Q (1 Q = 10^{18} Btu). It is instructive to compare this number with the total world reserve of coal which the United States Bureau of Mines has estimated as 68 Q, not all of it economically recoverable.

More to the point is the magnitude of the present rate of production. No official estimate has been published. Pre-war production potentials were about 1,000 tons of U_3O_8 per year (*11*). We can get a rough idea of the magnitude of the operations planned at Beaver Lodge, Canada, from figures published March 30, 1953 (*9*). It was disclosed that ore in the Gunnar lode which is a privately-owned zone 25 miles west of Eldorado's Ace mine, assays 4.5 lb U_3O_8 per ton, and that preliminary refining operations at Ace mill will be 500 tons or more a day at the start. Multiplying these figures together would give 400 tons of U_3O_8/year. With these numbers in mind, 1,000 tons/year of uranium does not sound like an unattainable production figure. We should be well satisfied if atomic power were supplying 10% of the country's power needs in 10 years; this would require the annual consumption of about 10 tons of nuclear fuel. It is patent that this could be done, using only 1% of the potentially available energy.

It is interesting to reflect that the present estimated world oil reserves are 0.4 Q, and that the United States annual consumption of oil is 0.01 Q, the amount of energy in the fission of 140 tons of uranium. The oil companies, the airplane engine and airframe manufacturers, and the automotive industry, to mention just a few, have built up sizeable enterprises on this negligible amount of energy. Although it would appear necessary ultimately to supplement the oil supply by hydrocarbon synthesis, in spite of repeated warnings by United States Department of Interior experts over the last 35 years that we are in imminent danger of running out of oil, it has not yet proved economic to make oil from coal.

In a similar way, there is nothing in the present fuel resource picture to compel us to await the design of breeding reactors before utilizing atomic energy for power; this argument for delay makes even less sense when we realize that in the meantime in our production reactors U^{235} is being consumed and the heat is being wasted. The argument has been advanced that 12 Q is not enough to sustain a stationary power industry in the year 2,000, and that we should, therefore, proceed now to develop breeder reactors which would use the more abundant U^{238}. The problem which we have to deal with, however, is not how to sustain a stationary power industry on uranium, but how to start one, and on this topic long-range planning considerations are unconvincing.

Conclusion

We see from the preceding sections that there is evidence that uranium will be abundant and cheap enough to be used as a fuel, and that it will not be necessary for the next several decades to go to extreme lengths to conserve it. Our studies of the economics of single-purpose stationary power plants show that reactors that consume uranium to a limited degree would be economic if they were built for less than $200 per kw. The key to commercial atomic power is a low-cost reactor. It is our belief that this objective can be achieved by inventive design with only minor advances to present technology.

* Recoverable uranium is defined as that obtainable at $100 lb or less for metal in billets.

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NEXT MONTH . . . .

"Selection of Materials and Equipment for Reactors"

A NUCLEONICS Special Report providing handbook-type information for all the components of different types of reactors. Included are sections on:

1. Materials. Shielding, structural, moderator, and reflector materials data are presented by George E. Evans, Oak Ridge National Laboratory.

2. Coolant. Coolant materials, pumps, blowers, piping, valves, etc., discussed by O. J. Woodruff, Jr., V. H. Hayden, and W. Purcell, Atomic Power Division, Westinghouse Electric Corp.

3. Reactor Operation. Electronic circuitry, radiation instrumentation, motors, magnets, safety and control rods, associated equipment are listed with desirable characteristics given by J. M. Harrer, Argonne National Laboratory.

4. Radiation Safety. Health-physics instrumentation and alarm systems are discussed by J. M. Smith, Jr., Hanford Works, General Electric Co.

5. Use of Reactor. Research on and with a reactor and special materials handling apparatus are discussed by M. Fox, Brookhaven National Laboratory.

6. Reactor Catalog Supplement. The compilation of reactor characteristics in the March, 1952 NUCLEONICS is brought up to date by H. S. Isbin, University of Minnesota.

Foreign Market for "Package" Nuclear Power

Appraisal of world power needs and costs indicates that there are a number of foreign countries that could profitably use small-capacity, high-cost nuclear power stations

By HOWARD C. GARY
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A SURVEY OF possible foreign markets for reactor power plants to supply 5,000 to 50,000 kw of electric power in the cost range from 2½ to 4¢/kwh indicates that there are areas in the world where such power could be competitive with existing power supplies or could supply needed power that does not exist now.

In a previous article,* a survey of U. S. domestic and overseas military markets for this type of power was described. For both these surveys, the same general principles apply; in the case of foreign markets, the general principles are particularly important since actual generating costs outside the U. S. are not available. An area that will find package nuclear power feasible must meet these four criteria:

1. It must be capable of using a firm flow of power. If not, the relatively high capital costs of the reactor would decisively give the advantage to lower-cost conventional facilities.

2. It must be fairly remote from cheap supplies of coal. Coal is a high-bulk, low-value item and, therefore, transport costs tend to be high.

3. It must be without access to reasonably priced hydroelectric facilities.

4. It must be unable to use oil or natural gas for reasonably priced power generation. It was shown in the first survey that unless diesel fuel costs more than about 25¢/gal, package nuclear power cannot possibly compete with diesel.

Unlikely Areas

With these criteria in mind, many areas can immediately be eliminated from consideration. Western Europe, for example, has adequate sources of power. The coal bottleneck in the early years after World War II has largely been removed. The area has been a big industrial center for so long that basic utility installations, including power, are in good order.

Other areas that can conveniently employ coal-burning steam plants include South Africa, most of eastern India, and south-eastern Australia.

Hydroelectric resources in central Africa, especially Nigeria and the Belgian Congo, appear adequate for all foreseeable power requirements. Certain parts of Chile and Brazil, as well as most of New Zealand, enjoy very satisfactory hydro potential. Uruguay also has reasonably adequate and well-located hydro resources.

The combination of adequate hydro resources and good supplies of low-cost coal appears to eliminate most of the Far East from further consideration.

Power requirements in most of the Middle East and along the northern African coast, as well as in Mexico, Colombia, Ecuador, Peru, and Venezuela, can be filled by relatively cheap supplies of diesel fuel and natural gas.

General Possibilities

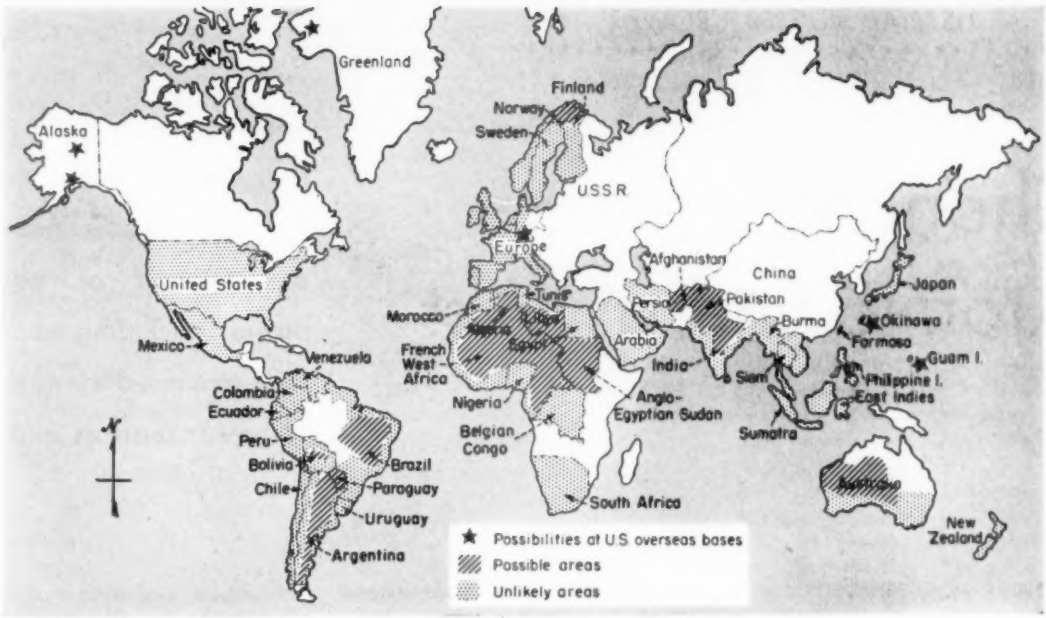
Despite the general availability of power in Western Europe, certain isolated areas, such as northern Norway and Finland, may be able to use the relatively high-cost package nuclear power.

In the Middle East, only Afghanistan does not have oil. In addition, this small mountainous nation has no coal. Preliminary investigation suggests that the capital city of Kabul may well be a potential site.

Certain inland portions of India and Pakistan are handicapped by their inability to produce reasonably priced power. Since both nations in the subcontinent hope to achieve a substantial degree of industrial development in the next few years, package nuclear power plants may be able to help fill their power void.

Large areas in north-central Africa—including the Sudan, French West Africa, Libya, and Algeria—are remote from coal, oil, and hydro resources. If the programs of technical assistance sponsored by the United States and the United Nations reveal the existence

*Richard W. Everett. Market for "package" nuclear power, NUCLEONICS No. 11, 4, 32 (1953).



POSSIBILITIES for economic use of high-cost package nuclear power

of a moderate industrial potential in this area, nuclear reactors may play a key role in the power development.

The area of greatest promise, however, seems to be the South American continent.

South America

The countries in South America that merit further consideration are Brazil, Chile, Argentina, Bolivia, and Paraguay. The rest of the Latin Republics have no problem because of their accessibility to plentiful supplies of oil.

Brazil. This country has tremendous hydro potential. Good water-power resources are located near the key industrial region of Sao Paulo. Other regions of this vast land that appear to have considerable industrial potential, however, are not so well endowed. In these spots, and the area north of the Volta Redonda is an especially prominent one, package power offers an exciting possibility.

An additional factor that enhances the attractiveness of nuclear power is the relatively high cost of imported diesel fuel at Brazilian ports. The cost hovers about 25¢/gal, which is just about the point where diesel power and package nuclear power might become competitive.

Chile. There is considerable hydro potential in this country also, except in the north, and the nitrate fields in that section of the country require a firm flow of power.

The cost of diesel fuel at the ports is a cent or two cheaper per gallon than the cost at Brazilian ports. But when the cost of transporting the fuel inland is considered, the price to users goes over the 25¢/gal mark. So, despite the abundance of hydro potential, there may well be an attractive market for package power in northern Chile.

Argentina. The large hydro potential in this country is in the foothills of the Andes, in the west, whereas the centers of population and industrial development are chiefly in the east. Imported diesel fuel costs over 25¢/gal.

It would appear likely, therefore, that many areas in Argentina offer attractive possibilities for package reactors. Whether the political situation there allows for real cooperation with the U. S. is, of course, a question.

Bolivia. Although a mountainous country, Bolivia's hydro potential does not seem suitably located to provide continuous and economic power to the many mines. Power experts have suggested that many mining installations in this land-locked country may be capable of making effective use of package power.

Paraguay. Lacking both coal and oil, and with hydro resources not conveniently located with respect to markets, this country offers promising possibilities. The fact that the railroads have wood-burning locomotives indicates Paraguay's power needs.

The Future

Further study and investigation may well turn up other isolated markets for relatively high-cost power. There may even be some good sites in nations where there is an adequate over-all supply of power. Fairly remote mining installations seem to be the type of project that can benefit most from package nuclear power.

The argument is frequently advanced that the high capital costs of investing in nuclear power would be a difficult barrier for most power-hungry nations to hurdle, the more so since the outlay would have to be in scarce—for them—dollar exchange. On the brighter side is the fact that this seems to be precisely the sort of project that can be financed with the help of the Export-Import Bank or the International Bank for Reconstruction and Development.

The first article in this series concluded that the outlook was not too promising for strictly commercial use of package reactors in the U. S. An appraisal of foreign markets, however, leads one to believe that there are several nations which might find package power an attractive solution to the power-shortage problem.—END

Changes in the Atomic Energy Act

Recommended here are four changes in the McMahon Act to foster industrial development of nuclear power—including changes to permit private ownership of power reactors and fuel

By OSCAR M. RUEBHAUSEN*

THE ATOMIC ENERGY ACT represents a unique departure from the traditional principles of our Government. The legislation was prepared by a nation deeply disturbed by, and deeply ignorant of, the force with which it must deal. Because of some misconceptions we adopted a legislative expedient which is inconsistent with the principles that have made us strong as a nation.

I think the time has come not only to take another look but to restore atomic energy to the normal framework of our democratic society.

Such a re-examination must precede any sound Congressional legislation in this field. Such a re-examination, moreover, is urgently needed if we are to drive forward with vigor and imagination to a maximum realization of the potentialities inherent in the atom.

Even now, however, it is possible to visualize the outlines of a new atomic policy. Four elements of such a new policy are reasonably clear and are tabulated on the next page. If a new atomic policy is to be founded, as I believe it should be, on these four new elements, an amendment of the Atomic Energy Act is clearly required.

Ownership of Facilities

Section 4 of the Atomic Energy Act provides that the United States shall be the exclusive owner of all facilities for the production of fissionable material except, to a limited extent, certain research facilities.

If there were to be a transfer of all atomic facilities to some international body, Government ownership of atomic facilities might still be wise. It would certainly simplify the legal steps for such a transfer. Since there is to be no such transfer, however, technical ownership in the United States Government serves no useful purpose which could not be as readily achieved in some way more consistent with our traditions. On the other hand, the elimination of this provision of the Act will not of itself bring about private participation in the atomic energy business. But, unless private ownership of facilities is permitted, private participation may be needlessly slowed up. Private ownership, it is said, will facilitate the financing of private atomic enterprises and will contribute to management efficiency. Repudiation of the principle of exclusive government ownership of atomic facilities will also, I believe, have a healthy effect on the national approach to atomic problems. It will shift some of the responsibility and, with it, the initiative to private hands.

Ownership of Fuel

Closely related to this question of the ownership of atomic facilities is the problem of who may own fissionable materials. In a sweeping provision, Section 5 of the Atomic Energy Act automatically vests in the Atomic Energy Commission all rights in any fissionable material which is within the jurisdiction of the United States.

Fissionable materials are clearly affected with the public interest. The public has a vital stake in their manufacture and in their use. This public interest can be served only by the very closest of governmental regulation. Such regulation should extend not only to matters of public health, safety and the national security, but also to the allocation, utilization and requisitioning of critical items. These are clearly difficult problems. They will require wise administration and flexible regulation.

Complex as these problems are, absolute government ownership of fissionable materials is not the answer. With ownership of fissionable materials goes the main initiative for forward progress in peaceful atomic uses. With it also go the burdens and the risks of failure. Such initiative and risk-taking are not the distinguishing characteristics of government—no matter how able and enlightened. Initiative and risk-taking are, however, the very essence of free private enterprise.

Control of Information

The section of the Atomic Energy Act which above all others, to my mind, urgently requires revision is Section 10. Section 10 defines what constitutes "restricted data." It gives the Commission control over the dissemination of such data and severely punishes any unauthorized or unlawful disposition of such data.

The key provision is the definition of restricted data; it is exceedingly broad.

We would expect restricted data to include all data concerning the manufacture or utilization of atomic weapons.

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Four Elements of a New Atomic Policy

1. Private ownership of atomic facilities should be permitted.
2. Private ownership of fissionable materials should be permitted but closely regulated by the Federal Government.
3. The definition of restricted data should be drastically revised and the role of secrecy as an affirmative instrument of national policy should be reduced.
4. Private patent rights in the atomic field should be recognized to the extent they do not involve military weapons.

The Act, however, goes much further. It also provides that restricted data shall include all data concerning "the production of fissionable material" and, what is more, "the use of fissionable material in the production of power." The only exception is for "data which the Commission determines may be published without adversely affecting the common defense and security."

Thus, all data in the atomic energy field is, you might say, born into this world secret; and it stays secret until the Commission feels that it can meet the burden of proving that the release of such data cannot adversely affect the common defense and security.

Such a definition is unduly restrictive. If interpreted literally, little could be released. For even the most innocuous information might, under certain circumstances, adversely affect the common defense and security.

Our basic atomic legislation, I think, should reflect a more balanced approach to secrecy. There should be more of a weighing of the total good that may result from publication as against the possible harm. There should be more emphasis on the affirmative policy of achievement as against the dubious gains from concealment. There should be some difference in treatment as between data pertaining to military applications of atomic energy and the nonmilitary processes. The legislation, moreover, should reflect that, in the last analysis, real security will come only if our rate of progress exceeds that of our enemies. Hence, control over atomic information should be directed above all else, not to the end of concealment, but to the assurance of our own maximum possible rate of progress in all phases.

Section 10 of the Atomic Energy Act should be revised. In this connection I make two specific suggestions.

First, I suggest that the definition of restricted data be limited only to data concerning the manufacture or utilization of atomic weapons and such other data as the Commission, from time to time, may affirmatively declare cannot be published without adversely affecting the common defense and security. Under this proposal data concerning the production of fissionable material or the use of fissionable material in the production of power would become public unless the Commission found positive reasons to the contrary. The burden of proof would be completely reversed.

A change in the definition of restricted data would not, of itself, be enough. Habits formed in the six years of operation under the present law may be hard to break. So my *second* suggestion would be, once the definition was changed, that the Commission appoint a special panel to review all secret data in the light of the new policy. Such a panel should have a specific mandate from the Commission to declassify all information which is not in the weapons field unless it is satisfied that such declassification would adversely affect the common defense and security. So much for Section 10.

Patents

The fourth principal area for legislative attention is Section 11, patents.

As the Act now stands no patent shall be granted for any invention or discovery which is useful solely in the utilization of fissionable material or atomic energy for a military weapon, and no patent shall confer any rights to the extent any invention or discovery is used in the utilization of fissionable material or atomic energy for a military weapon. With such provisions I personally cannot disagree.

The Act, however, goes further. It provides that no patent shall hereafter be granted for any invention or discovery which is useful solely in the production of fissionable material. Likewise, it denies patent rights with respect to any invention or discovery to the extent that it is used in the production of fissionable material. I do not see what is to be gained by such a repudiation of the incentives and benefits of the patent system in the case of fissionable material for other than military purposes.

If inventions or discoveries have uses which are not military or governmental, then the rights to such private uses should be enjoyed by those who have contributed to their discovery.

If the government makes a discovery through government-financed research, there should, clearly, be no private patent rights. If, on the other hand, the know-how or the capital of private industry is drawn upon in the making of such a discovery, then the contribution of private industry to that discovery should be recognized in the distribution of the patentrights.

The form of such recognition will, of course, vary in every case. What is fair will depend on a wide variety of factors. Such matters are, however, subject to negotiation and reasonable allocation. It is only with the principle of fair recognition for the contributions of private persons that I am presently concerned. To the extent that the Atomic Energy Act repudiates this fair principle in the nonmilitary and nongovernmental uses of atomic energy, I believe it should be amended.

Only with a change of policy can we bring into play the full vitality of the American people and American industry. We will need every bit of that vitality in the days ahead.

* * *

The article is based on a paper presented before the American Power Conference in Chicago, Illinois, March 25, 1953, to be published in full in the May issue of the New York Bar Association Record.

Recent Advances in Reactor Technology

The problems of breeding and of achieving high flux are treated in detail in this description of reactor developments. The Homogeneous Reactor Experiment, the Materials Testing Reactor, and others are described in relation to these problems

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THE TECHNOLOGY of reactors has progressed considerably in the last couple of years. Much of this progress is in areas covered by security regulations. But even leaving these areas out, there is a great deal to say about the current crop of reactor experiments and of experimental reactors.

I use the two distinct terms "experimental reactor" and "reactor experiment" advisedly, for, as the technology unfolds, it has become more and more clear that a distinction exists between reactors built as pilot plants (reactor experiments) and reactors built as neutron sources for general scientific exploitation (experimental reactors).

That pilot plants are necessary in a technology as bizarre as nuclear energy might seem to be a commonplace, but partly because the Hanford reactors were successfully built essentially without a pilot plant, and partly because a deeply essential characteristic of a reactor—its power per unit volume—can be realized only in a full-scale system, it had been customary to circumvent small prototype reactors in proceeding to a full-scale system. This trend has, I think fortunately, been reversed, and two pilot plant reactor experiments, the Experimental Breeder Reactor and the Homogeneous Reactor Experiment, have been constructed in the last two years. Both these reactors, which are in the 1,000-kw class, are much smaller than their ultimate embodiment. Although their small size makes them incomplete pilot plants, they do have enough in common with big-scale breeders or homogeneous reactors to make the experience gained with them enormously relevant.

Experimental Breeder

EBR has already been described in considerable detail.* It is a pilot plant for a fast-neutron breeder, and therefore contains no moderator and is, in consequence, extremely small. That a breeder should operate on fast-neutron fission rather than on slow-neutron fission is suggested by the manner in which the fission-to-capture ratio in U^{235} varies with energy.

Capture of a thermal neutron by U^{235} leads to fission (with a yield of 2.5 neutrons per fission) about 83% of the time and to radiative capture (with a yield of no neutrons) the remaining times. Thus when a U^{235} nucleus is destroyed by thermal fission, only about 2.14 new neutrons are released. For fast neutrons, η, the number of neutrons produced per U^{235} destroyed, is substantially larger than 2.14. The feasibility of breeding thus depends very sensitively on $\eta - 2$, the breeding gain, since one neutron is needed to maintain the chain and one neutron is needed to make a new fissionable nucleus to replace each destroyed one; and it is natural to look toward fast reactors which have high breeding gain for success in breeding.

EBR, being nonmoderated and therefore very small, requires the very best heat transfer medium if specific powers of reasonable magnitude are to be achieved. The actual coolant used in EBR is NaK eutectic which melts at 12° F and boils at 1,500° F. The operating temperature of EBR, 660° F, is rather modest but is sufficiently high to produce 400-lb steam. This steam is now being used routinely to produce about 150 kw of electricity—the first electricity from nuclear fission.

Homogeneous Experiment

The other reactor experiment which is just getting underway is HRE at Oak Ridge. This system consists of an aqueous uranyl solution which operates at sufficiently high temperature, and therefore pressure, to produce steam which runs a small turbine.

The reactor is water-moderated and is therefore a thermal reactor. As a potential breeder, with D_2O replacing the H_2O, it would suffer by comparison with the nonmoderated EBR; the value of η for U^{235} would be only about

* W. H. Zinn, NUCLEONICS **10**, No. 9, 8 (1952).

2.14. However, the heat produced in this reactor is transferred by actually circulating the fluid fuel round and round through an external heat exchanger, and this relatively efficient heat transfer method makes possible much higher specific power output, i.e., power per gram of uranium, than is attainable in EBR. Thus the turnover rate of the fuel is potentially much higher than in a fast breeder, and it is hoped that the yearly interest rate on the fuel, which is proportional to specific power times breeding gain, might be competitive with the yearly interest rate in a fast breeder, which has higher breeding gain but lower specific power.

Perhaps the most interesting physical questions in an aqueous circulating fuel system are connected with its stability. By virtue of its fuel circulation, such a reactor loses many of its delayed neutrons. Since the delayed neutrons tend to lengthen the time scale of the reactor, it would be supposed that their loss might seriously compromise the stability of the system against small fluctuations. Although, in general, this is true, this effect is balanced in an aqueous system operating at high temperature by the fall in density and in H_2 scattering cross section as the temperature rises. On both these counts the leakage of neutrons from the system increases as the temperature goes up, and this gives the reactor a strongly negative temperature coefficient, as much as $-10^{-3}/^\circ$ C.

The kinetics of such a "no-delayed-neutron" reactor into which an increment of reactivity, δC, is suddenly introduced, is easily deduced, as shown by J. Stein (Westinghouse), from the reactor's "equations of motion." These are, for power removal at a constant rate, P_0,

$$dQ/dt = SdT/dt = P - P_0$$
$$dP/dt = [(\delta C + \alpha T)/l]P$$

where Q is the heat content, S is the heat capacity of the reactor, α is the temperature coefficient of reactivity, l is the generation time, and P is the power at time t. This pair of nonlinear equations can be discussed most easily by writing the differential equation of the phase diagram

$$dP/SdT = [(\delta C + \alpha T)/l(P - P_0)]P$$

which, on integration, gives

$$\frac{P - P_0}{P_0} - \ln\frac{P}{P_0} = \frac{S}{P_0 l}\left(T\delta C + \frac{\alpha}{2}T^2\right)$$

The reactor is seen to undergo relaxation oscillations, provided α is negative; if α is positive, the reactor is unstable. The maximum temperature excursion is

$$T_{max} = -2\delta C_0/\alpha \qquad (1)$$

This excursion is just twice the excursion, $|\delta C/\alpha|$, which would reduce the reactor to the just-critical state. Equation 1 means that even if 1% excess reactivity is inserted into a reactor which has *no* delayed neutrons, but for which $\alpha = -10^{-2}/^\circ$ C, the maximum temperature rise is only about 20° C.

The delayed neutrons damp the oscillations. This is seen most easily for the case of small oscillations. Without delayed neutrons these oscillations are sinusoidal with period

$$t = 2\pi\sqrt{lS/|\alpha|P_0}$$

The delayed neutrons damp the oscillations with a damping constant $2l/\beta$, where β is the fraction of delayed neutrons.

Thus, as long as the temperature coefficient is negative and large, circulating fuel systems seem to be stable; or, at least, oscillate with finite amplitude. Because of the very strong coupling between reactivity and temperature, such reactors are inherently self-regulating. Any increase in power demand (as for example, opening the steam throttle, i.e., cooling more efficiently) tends to cool the inlet fluid to the reactor; this increases the reactivity, which increases the power. This raises the outlet temperature until the extra reactivity is balanced by a restoration of the original temperature profile. To change the temperature of such a strongly coupled system, only the control rod need be moved. If the reactivity is reduced, the system temperature will fall until the loss in reactivity is restored by the lower temperature.

Experimental Reactors

With this very quick resumé of the status of our reactor experiments I now turn to the new experimental reactors—the neutron sources. In this country we are particularly fortunate for possessing U^{235}. Since enriched reactors are small we can, on this account, make extremely intense reactor neutron sources with relatively modest-scale devices. This philosophy underlies the Los Alamos and North Carolina State College water boilers, both of which run at less than 50 kw yet achieve a flux comparable to the flux in the huge Oak Ridge or Harwell graphite reactors. It also underlies the most advanced experimental reactor ever built—the Materials Testing Reactor—as well as its daughter, the Low Intensity Test Reactor.

MTR was designed and built as a joint enterprise of the Oak Ridge National Laboratory and the Argonne National Laboratory. It is located at the Reactor Testing Station, Arco, Idaho, where it is operated by the Atomic Energy Division of the Phillips Petroleum Company. It is now the major irradiation facility of the Atomic Energy Commission. LITR was originally a hydraulic mockup of the MTR and was converted into a nuclear reactor at ORNL in 1950.

In designing a reactor radiation source it is desirable to maximize both the fast-neutron (> 1 Mev) and the thermal-neutron flux. If maximum slow-neutron flux is desired, then, since power output in a reactor is proportional to fission rate,

$$P = N\sigma_f\Phi_{th}$$

or the thermal flux, Φ_{th}, is

$$\Phi_{th} \sim P/N$$

where N is the number of U^{235} atoms per cubic centimeter, σ_f is the thermal fission cross section, and P is power per cubic centimeter. Thus the thermal flux is proportional to the power per kilogram of uranium.

To achieve maximum thermal flux it is therefore necessary to maximize the heat transfer from each gram of uranium fuel. This is automatically accomplished in a homogeneous reactor, and it implies in a heterogeneous reactor that cooling *surface* per gram of uranium be as large as possible.

To maximize the virgin or uncollided flux it is necessary to maximize the power per unit volume of reactor. The uncollided flux of neutrons, Φ_0, is

$$\Phi_0\Sigma_s = q$$

where Σ_s is the macroscopic scattering cross section; q is the number of neutrons produced per cubic centimeter per second and is therefore proportional to the power per unit volume, P. Thus

$$\Phi_0 \sim P/\Sigma_s$$

or, since Σ_s at high energy is rather insensitive to the kind of moderator, the virgin flux is proportional to the power per unit volume.

Thus, to build an experimental

reactor which maximizes both the slow and the virgin fast-neutron flux it is desirable to extract as much heat as possible from a given amount of uranium, and at the same time to keep the reactor volume as small as possible. Both conditions are met automatically in an aqueous homogeneous reactor. However, the chemical engineering problems involved in running aqueous systems at high nuclear power are so formidable that it was deemed best to make such a maximal neutron source with a heterogeneous reactor.

One solution to the problem of designing a heterogeneous reactor as a maximal neutron source is that used in the Swimming Pool Reactor.* In this reactor, the heat transfer area per gram of uranium is maximized by disposing the U^{235} in a thin plate; volume is minimized by using the same water for moderating and cooling. A stack of 20 fuel elements is immersed in a pool of water. The fuel elements consist of thin parallel plates of aluminum sandwiches with an enriched uranium filler as the "meat" of the sandwich. The plates are spaced at intervals of a few millimeters, and this space is used by the water both to cool and to moderate. The control elements consist simply of cadmium rods which move up and down in unused fuel-element spaces.

A reactor of this sort is extremely flexible; many different fuel loadings are permissible, and in this way the intensity and the spectrum of the flux at the experimental ports can be changed at will. The Swimming Pool Reactor is cooled only by natural convection. It normally operates at about 100 kw, at which power, since the critical mass is about 3 kg of U^{235}, the average slow neutron flux is about 10^{12} n/cm²/sec.

LITR and MTR are also heterogeneous, light-water cooled and moderated, enriched-uranium reactors designed for maximum neutron source characteristics. LITR provides an average thermal neutron flux greater than 10^{13} n/cm²/sec, whereas MTR provides an average thermal neutron flux greater than 10^{14} n/cm²/sec. In MTR the maximum neutron intensity available in a beam hole is over twice the average flux. Both reactors have power levels measured in megawatts.

I suppose it is fair to say at this time that one of the pleasantest surprises of the post-war nuclear energy development has been the extraordinary reliability with which MTR operates. This is particularly significant since it demonstrates that very high power density—hundreds of kilowatts per liter for MTR—in itself is not prejudicial to the success of a nuclear reactor.

There are two nuclear characteristics of MTR—the shutdown behavior of the reactor and the neutron buildup in the reflector—which are of interest. As is fairly well known by now, Xe^{135}, a 6% yield fission-product daughter of the 6.7-hr I^{135}, has a slow-neutron cross section of 3.5×10^6 barns. In a reactor running at a flux of 10^{14} n/cm²/sec, the half-life of Xe against neutron capture is about $\frac{1}{2}$ hr, whereas, the natural lifetime of Xe^{135} against beta decay is about 9.2 hr. Thus in MTR, nearly every Xe^{135} atom formed absorbs a neutron and the reactivity of MTR gradually decreases as the Xe^{135} builds up by decay of its parent I^{135}. This reactivity loss saturates at about 4.5%, this being the product of the fission yield by the fraction of neutrons which go into fission.

After the reactor is shut down, the I^{135} continues to produce highly absorbing Xe^{135} atoms; however, because there are now no neutrons to destroy the Xe^{135} its concentration will rise and will continue to rise until the natural decay of the Xe^{135} overbalances the decay of the I^{135}. Thus a reactor whose thermal flux is 10^{14} n/cm²/sec will undergo a great decrease in reactivity after shutdown, and after about 10 hr, when the Xe^{135} concentration is at its maximum, the reactor will have lost as much as 20% in reactivity. This means that a few hours after a high-powered reactor such as MTR has been shut down, it is impossible to start it up again unless considerable excess reactivity is built into it, or unless new fuel which contains no Xe^{135} is placed in the reactor, or unless sufficient time is allowed for the Xe^{135} to decay.

The other design characteristic of these small thermal reactors has to do with a method of artificially increasing the thermal-neutron flux in the reflector which surrounds the experimental beam holes. Since the lifetime against capture of a thermal neutron in a Be or BeO reflector is very much longer than it is in the highly absorbing reactor, the thermal-neutron density tends to build up in a slightly absorbing reflector. In this way it is possible to increase the thermal flux in some parts of the reflector of MTR. This of course is valuable to experimenters who need the highest flux in the region outside the reactor.

Commercial Power

This account of our reactor experiments and of our experimental reactors would be incomplete, as is any article on atomic energy, without some observations on why we don't now have commercial atomic power on a big scale. To add my ideas to all the others that have sought to answer this complicated question, I shall make only this technical, not political, observation: commercial power technology is basically a conservative technology, it depends for its economic success on small profits in large units over very long times. Thus, successful commercial steam prime movers must be devices which last a long time, say 20 yr, whereas the only *experimental* nuclear electrical producers, EBR and HRE have produced electricity for only one year, and less. The lifetime of any high-temperature device is limited by the integrity of the materials. Even without radiation, prediction of how long any such device can last is at best risky, and there is no substitute for time in getting a reliable estimate of longevity. So it is that one of the central reasons we can go only so fast in achieving successful large-scale commercial nuclear power is that we cannot predict with confidence how long our nuclear-producing gadgetry will last. Questions like this naturally lie in the province of the solid state physicist and metallurgist.

With magnificent radiation sources such as MTR available, I think that we can now embark on the time-consuming, but ever so necessary job of finding out, instead of guessing, how long we can expect materials to hold up in the strange and exacting environment which they encounter in a high-temperature, high-radiation reactor. Only when adequate answers to these questions are available can we hope to achieve a full-scale nuclear power enterprise that is competitive with ordinary power sources.

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This article is based on a talk given at the North Carolina meeting of the American Physical Society, March 27, 1953.

* William M. Breazeale, NUCLEONICS **10**, No. 11, 56 (1952).

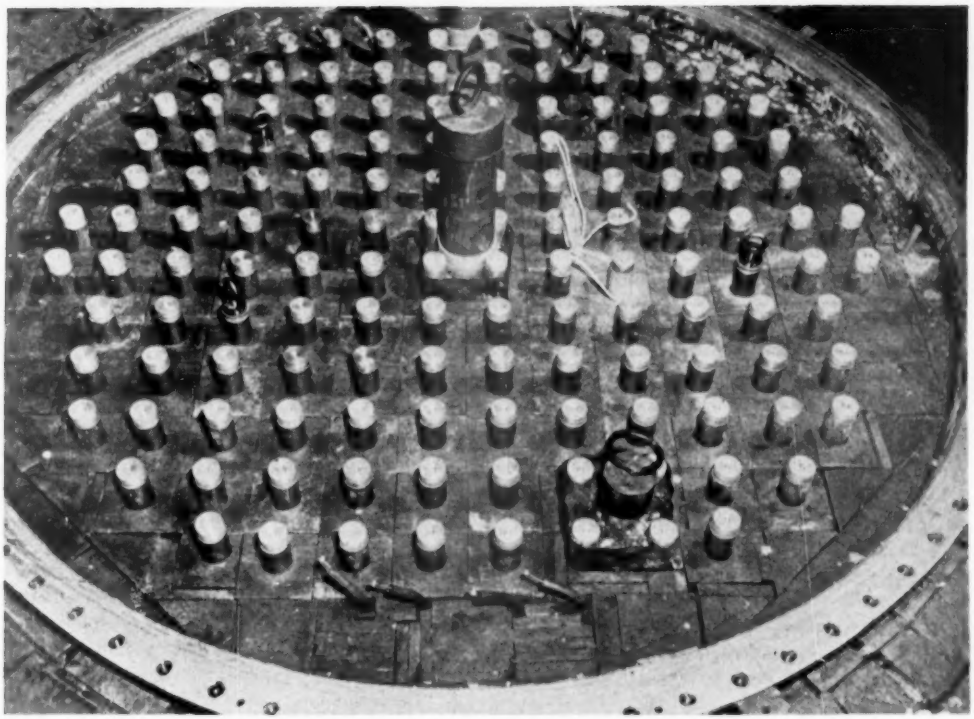

FIG. 1. Uranium rods suspended in heavy water in Argonne reactor

Heavy Water in Nuclear Reactors

The nuclear properties of heavy water make it valuable as a moderator for thermal reactors. The behavior of heavy water in reactors, special handling procedures, and quality checks are discussed, with particular reference to CP-3′ reactor

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ALTHOUGH THERE ARE definite differences between the physical and chemical properties of heavy and normal water, as shown in the table (*1, 2*), these are not so marked that they are noticeable from a casual examination of the two liquids. It is the difference in the nuclear properties that makes heavy water of value in nuclear reactors.

The scarcity of heavy water (about 0.015% of normal water) and its physical characteristics require that special precautions be taken in preparation and handling to obtain it in high purity and to maintain the purity under operating conditions in a reactor.

The slight differences in physical properties of H_2O and D_2O, and of other compounds of hydrogen and deuterium, form the bases of several methods for separating the isotopes and producing heavy water.

One of the earliest methods employed in the preparation of heavy water made use of the difference in the ease with which hydrogen and deuterium may be discharged at the cathode of an electrolytic cell. Fractional distillation of water through several stages also has been used to separate heavy water from normal water. A combination of electrolysis and an exchange method (*3*), whereby deuterium, mixed with hydrogen in the gas phase, may be exchanged with the hydrogen of water in the liquid phase to enrich the deuterium in the water, provides a successful process for the production of heavy water. Use of the proper catalyst to facilitate the exchange, and cascade operation of the system results in a good yield of high-purity heavy water.

Nuclear Properties

The production of high-purity heavy water is a quite complicated and expensive process which might not be

justified for the purpose of supplying heavy water for reactors unless there were more important differences between D_2O and H_2O than are evident in the physical and chemical properties so far mentioned. The research and investigation performed since the discovery of the neutron (*4*) in 1932 have established such differences (*5–8*) in the nuclear properties of hydrogen and deuterium; these differences are sufficiently great that they are of important consideration in nuclear reactors.

The thermal-neutron scattering and absorption cross sections (*6, 9*) for the hydrogen atoms of water are, respectively, about 50×10^{-24} and 0.33×10^{-24} cm². The corresponding values for the deuterium atoms of heavy water (*7, 9*) are about 7×10^{-24} and 0.00046×10^{-24} cm². From these values it is evident that the hydrogen atoms of light water are more effective in scattering neutrons than are the deuterium atoms of heavy water. However, the much higher absorption cross section of hydrogen for thermal neutrons causes normal water to be far inferior to heavy water as a moderator in a reactor where neutron economy is a prime consideration. This is evident from a consideration of the moderating ratios (*10*) of the two liquids. If the average logarithmic decrement of energy of neutrons upon collision with a nucleus is ξ and the macroscopic absorption and scattering cross sections, respectively, are Σ_a and Σ_s, then the moderating ratio is $\xi\Sigma_s/\Sigma_a$. For normal and heavy water, the moderating ratios are, respectively, about 72 and 12,000. Thus, normal water is about 0.006 as desirable as heavy water from the consideration of the moderating ratios. With normal uranium fuel elements, it does not appear at all possible to develop a nuclear chain reactor using normal water for the moderator.

The effects of the nuclear properties of the moderator on the dimensions of a reactor and the spacing of the fuel elements in a heterogeneous reactor are probably more evident in terms of such quantities as the diffusion length, L, and the Fermi age, τ, or the slowing-down length $\tau^{1/2}$.

The diffusion length is the square root of one-sixth of the mean-square distance that a neutron, after becoming thermal, diffuses through a medium until it is captured in the medium. The slowing-down length is the square root of one-sixth of the mean-square distance a neutron travels in a medium while it is being slowed down from the original energy possessed at its formation by fission until it has reached thermal energy. These quantities thus give indications of the dimensions required in reactors for the moderators to perform their functions.

Physical Properties of Light and Heavy Water

Property	H_2O	D_2O
Freezing point (° C)	0	3.8
Boiling point (° C)	100	101.4
Temperature of maximum density (° C)	3.98	11.23
Maximum density (gm/ml)	1.000	1.106
Heat of vaporization at 25° C (cal/mole)	10,515	10,846
Heat of fusion (cal/mole)	1,436	1,515
Viscosity at 5° C (cp)	1.519	1.988
Viscosity at 35° C (cp)	0.721	0.864
Vapor pressure at 10° C (mm Hg)	9.209	7.79
Vapor pressure at 90° C (mm Hg)	525.72	495.5

For normal water, the values (*10*) of L and $\tau^{1/2}$ are, respectively, about 2.88 and 5.7 cm; the corresponding values for heavy water are about 100 and 11 cm.

Thermal Reactors

When a moderator is mentioned as a component of a reactor, it is recognized that the reactor operates chiefly on thermal neutrons. This is another way of saying that the primary function of a moderator is to affect fission neutrons, or neutrons of high energy, in such a way that they are reduced in energy down to the thermal region, where they are much more effective in causing fission of U^{235}.

A characteristic which has come to be associated with moderators is low atomic mass. Probably the chief reason for this is the necessity for having the neutrons reduced to thermal energy by a few collisions in the moderator. This requirement is imposed by resonances in the nonfission absorption cross section of U^{238} in the neutron energy range between fission and thermal and also, of course, by increased loss of neutrons through leakage from the reactor and absorption in the moderator and structural materials of the core as the number of slowing-down collisions increases.

Thus, the function of a moderator in a natural uranium reactor is to reduce rapidly to thermal the energy of the neutrons created in fission, i.e., with few collisions and little loss of neutrons in the moderator itself. A further function of the moderator is to provide uranium-free regions where the moderation of the neutrons can be accomplished without more than a few neutrons being able to encounter U^{238} while in the energy range between fission and thermal.

Figure 1 shows the top of the core of the Argonne heavy-water reactor. The shield has been removed to show the ends of uranium rods suspended in heavy water.

Since the neutron economy balance is very critical in a natural uranium reactor, it is not possible to establish a chain reaction in such a machine without having the moderator present. One of the advantages of a liquid-moderated reactor is the ease with which it may be secured against inadvertent operation. This may be accomplished by simply draining or dumping the moderator. Hence, a storage tank for the moderator is characteristic of such reactors if the moderator needs to be saved because of its expense or radioactivity content. The smallness of the per cent of U^{235} present in normal uranium, about one part in 140, or 0.7%, means that high-purity heavy water must be used if a liquid-moderated reactor is to be considered.

When uranium enriched in U^{235} is available for fuel, either as a heterogeneous arrangement or as a homogeneous mixture of fuel and moderator, the requirements imposed upon the moderator are not so stringent, unless perhaps neutron economy is the dominant factor to be considered. Thus, in general, heavy water is required and used as a liquid moderator with normal uranium, and normal water may be used with uranium that is sufficiently enriched in U^{235}. The enrichment and the amount of the U^{235} required for a reactor using normal water will usually be greater than for a reactor employing heavy water as the moderator. This will be dependent upon the quantities of various materials which may enter into the design and construction of the reactors.

As would be expected, the optimum

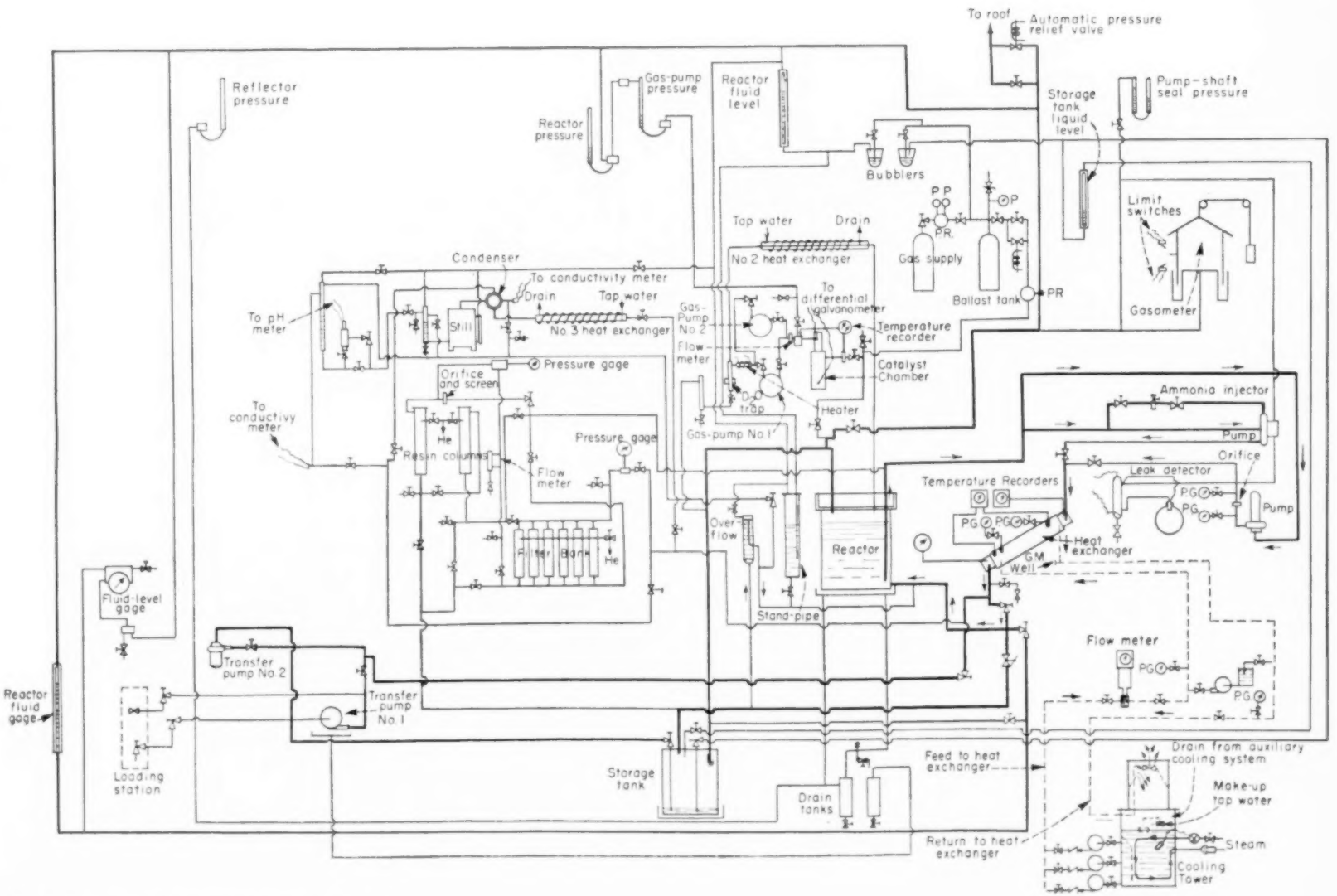

FIG. 2. Flow sheet of the heavy-water system of CP-3

spacing or geometrical arrangement of the fuel elements in a heavy-water reactor is somewhat more generous than if ordinary water were used. This feature is indicated by the relative diffusion lengths and Fermi ages for thermal neutrons in normal and in heavy water.

It is also possible, at a given power level, to have more neutrons available for research or experimental purposes and for other useful production from a heavy-water reactor than from one using ordinary water. Such considerations as these have continued to focus attention on heavy-water reactors, even though it is now possible to produce uranium which is highly enriched in U^{235}.

Moderator and Coolant

In addition to requiring a moderator for the operation of a thermal-neutron reactor, a means must be provided for removing the heat released in the fission process. For all reactors, except those operated at low power, the coolant must function chiefly by convective or circulatory effects. As the specific power (the power produced per unit mass of the fuel) and, of course, the power per unit volume of the core of the reactor, is increased, the problem of heat removal becomes correspondingly more difficult. Since the thermal capacities and viscosities of ordinary water and heavy water are similar, either may be used successfully as a coolant or heat-transfer medium in a thermal-neutron reactor. However, the greater neutron absorption of normal water either restricts its use to reactors supplied with enriched uranium or requires that normal water constitutes only a small volume of the liquid used in the core of the reactor.

Heavy-Water Reactors

There are two types of heavy-water reactors in operation at the present time. The CP-3′ reactor at the Argonne National Laboratory, which was the world's first heavy-water reactor, is representative of a group of low- or medium-power reactors which employ heavy water as a dual-purpose moderating and cooling medium. These reactors, which include those developed and operated in the United States, Canada, France, Norway, and one soon to start operation in Sweden, circulate the heavy water either by convection in the reactor tank or, at relatively low flow rates, through heat exchangers. By these circulation processes, heat is transferred from the fuel elements to the external surroundings.

The NRX reactor in Canada (*11*), representing the second type, normally operates at a power in excess of 10 Mw and is cooled chiefly by circulating ordinary river water through tubes in the reactor tank. The tubes serve to keep the heavy water and the ordinary water separated and provide thin annular channels surrounding the fuel rods, past which the ordinary water moves with sufficient speed to carry away the heat developed in the fuel rods.

Originally, all these heavy-water reactors were loaded with natural uranium fuel. For this reason, the amount of light water which could be present to circulate through the core, as in NRX, had to be kept to a small volume. In 1950, CP-3′ was supplied with fuel rods made from enriched uranium alloyed with aluminum. The neutron economy attained by the reduced amount of U^{238} in the fuel rods and the low absorption of neutrons in the heavy-water moderator-coolant made available for experimental purposes a maximum thermal flux of about 3×10^{12} to 4×10^{12} n/cm²/sec at power levels somewhat less than 300 kw. High neutron fluxes are obtained in CP-3′ with the use of considerably less U^{235} and lower power levels than would be required if ordinary water had been used as the moderator-coolant.

Behavior in Reactors

The small differences between the viscosities, densities, and vapor pressures of light and heavy water do not require different types of pumps, heat exchangers, or other equipment for use in the circulation of the two liquids. At very high flow rates, and when it is desired to push a heat removal system to its maximum efficiency, differences in design would probably be important.

A flow sheet of the heavy-water system of CP-3′ is shown in Fig. 2.

The chief considerations in most installations are probably given to preventing loss of heavy water from the system and preventing contamination of the heavy water with ordinary water from the secondary or other cooling system, and with water vapor from the surrounding air.

Sealed systems are demanded for heavy-water reactors in view of the above and also because nitrogen from the air would react with either water vapor or heavy-water vapor to form nitric acid in the presence of the radiation produced by a reactor. Special valve-stem and shaft seals, and canned types of motors and pumps, along with slight pressurization of the interior of a reactor with an inert atmosphere of a gas such as helium, provide means for keeping water vapor and air out of the core and the D_2O system. Many of the required pieces of equipment have been developed for other applications in industry and may be used, directly or with slight modifications, in the heavy-water reactor systems.

In addition to bringing about nitric acid formation, the radiation in a reactor causes a slight dissociation of the heavy water into deuterium and oxygen. To prevent loss of these constituents, as well as to eliminate the possibility of an explosive mixture, the gas atmosphere is circulated through a catalyst such as might be used by industry for the combining of hydrogen and oxygen. The effect of the catalyst is to recombine the deuterium and oxygen to form heavy-water vapor, with an evolution of heat occurring in the process. Measurements of this heat by means of thermocouples, flow-rate meters, etc. provides a means for monitoring this aspect of the reactor's behavior. In general, the dissociation rate of the heavy water is small, and the equilibrium concentrations of deuterium and oxygen in the reactor atmosphere remain low.

Since corrosion effects are associated with valence or ionic states, one would expect little difference between corrosion of a reactor's components by ordinary and heavy water. The exceptions to this would probably come as a result of radioactivity effects, which might be initiated by nuclear reactions. At the present relatively low neutron flux levels, any such effects do not appear very significant.

Special Handling

The naturally occurring small percentage of heavy water in ordinary water suggests that exchange of light-water vapor and heavy-water vapor would occur quite readily if heavy water were left exposed to the humid air of our normal atmosphere. There is also the likelihood of exchange

occurring between deuterium and hydrogen, particularly in the vapor phase of heavy water. Considerable care must therefore be exercised in storing heavy water and in transferring it from one container to another.

Rigid precautions must also be taken in circulating the heavy water in the reactor system to prevent mixing the normal humid atmosphere outside the reactor with the heavy water. Although nitric-acid formation seems to occur only in the regions of intense reactor radiations, the exchange of the light and heavy water vapors may occur equally well in all parts of the heavy-water system. Diffusion of the outside atmosphere through small openings into the slightly pressurized atmosphere of the reactor does not appear to be anything more than a remote possibility. But, of course, a loss of heavy-water vapor will occur to some extent whenever such small openings or leaks are present.

Because of the possibilities of dilution of the heavy water with light water through exchange in the vapor phase, as well as directly through failures or leaks in such equipment as heat exchangers, certain precautions and procedures have been developed in connection with the operation of the heavy-water reactor at the Argonne National Laboratory.

All heavy-water storage is provided in sealed drums or other containers of 28 aluminum or stainless steel. After each opening of the containers, they are purged of air by use of dry helium, and again sealed. Similar precautions are taken during the transfer of heavy water from a container to the reactor to prevent air from getting into the D_2O system. The use of helium might not be necessary, if dry air were supplied, except that the amount of nitrogen transferred to the reactor system would be increased by such a procedure and increased purging of the reactor atmosphere would be required.

To further guard against the contamination of heavy water with ordinary water, the close association of the heavy-water and light-water systems is limited to only those places which appear to be absolutely necessary. Certain heat exchangers are required which employ ordinary water as the cooling medium in one circuit. These heat exchangers are made as failure proof as possible. In the case of the main heat exchanger, used to transfer the reactor's normal 275 kw of heat to a cooling tower, operation is arranged so that the pressure in the heavy-water side of the heat exchanger is a few pounds per square inch higher than in the ordinary-water side. This provision assures that, for the occurrence of small leaks, some heavy water would be lost rather than some ordinary water added to the D_2O system. A G-M counter in the ordinary-water cooling line is arranged to monitor the circulation system for such leaks during all times of operation of the reactor.

The other matter of chief concern in connection with heavy water is actual loss through leaks in the circulation system, or loss of vapor through imperfect seals associated with the reactor atmosphere system. To guard against such losses, various types of leak detectors are employed which will provide alarms and visual indications of even very minute leaks.

Another caution, which should be taken in connection with the heat exchanger and cooling system, is to guarantee that circulation of light water can never occur at temperatures below about 5° C. Since heavy water freezes at 3.8° C, it is possible to cause damage, directly or otherwise, through freezing of the heavy water by circulating ordinary water which is too cold.

Quality Checks

Such routine checks of the quality of heavy water as given by electrical conductivity and *p*H or *p*D tests, which are common practice on process liquids for industry, are also the usual ones made, by recording instruments, on the heavy water of a reactor. More precise determinations of these properties are made at regular intervals by laboratory-type instruments and sampling techniques.

More specialized checks of such characteristics as the amounts, in parts per million, of metallic ions present in the heavy water and the isotopic purity of the heavy water are made by optical spectroscopic and mass spectrometric methods.

Various characteristics such as density, index of refraction, etc. may be used as a measure of the D_2O percentage of the heavy water. However, mass spectrometric determination of the D_2O purity is considered the most reliable. The technique employed (*2, 12*) at Argonne makes use of the exchange equilibrium between the gases D_2, HD, and H_2, which can take place in the vapor phase in the presence of a suitable catalyst, to provide a gas sample which may be analyzed on a mass spectrometer to determine the isotopic purity of a D_2O sample.

Other mass spectrometer techniques may be employed to monitor the isotopic purity of the heavy water in an operating reactor. However, the periodic sampling technique appears more satisfactory at the present time, since other relatively simple recording instruments for *p*D, electrical conductivity, etc. may be strategically located to indicate any rapid deterioration of the heavy water quality.

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Neutron Balance in Reactors

The most important characteristic of a reactor is its neutron balance. A typical balance sheet for a graphite-moderated reactor is:

Neutrons from U^{235} and fast fission	2.56
Neutrons carrying on chain reaction	1
Neutrons absorbed in U^{238} to make Pu^{239}	0.9
Neutrons absorbed in U^{235} to make U^{236}	0.2
Neutrons absorbed in moderator	0.3
Neutrons absorbed in structural materials	0.05
Neutrons escaping from core	0.09
Excess neutrons	0.02

—From *Nature* **171,** 411 (1953)

Apparatus for Large-Scale Production of Phosphorus-32

With the extraction and purification process described, 2–3 curies of phosphorus-32 have been obtained from 1.8-kg batches of sulfur irradiated in the flux in BEPO at Harwell. The two-day procedure gives a 60–70% yield without carrier

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THE APPARATUS (*1, 2*) used here for the routine production of phosphorus-32 in activities of up to 2 or 3 curies per batch has proved satisfactory in use over the last two years.

The chemical production process used differs very little from that developed at Oak Ridge. Sulfur is cast into 300-gm blocks which are irradiated for two weeks in the highest available flux in BEPO, the more powerful of the two graphite piles at Harwell. The (n,p) process on the S^{32} of the normal mixture of sulfur isotopes results in about 1 mc of P^{32} being formed per gram of target material. The only other important activity is S^{35} from the reaction $S^{34}(n,\gamma)S^{35}$. The activity of S^{35} per gram of sulfur is about 2–300 μc.

Phosphorus-32 is extracted and purified in two processes. In the extraction, the irradiated sulfur is melted and agitated under pressure with dilute nitric acid. Purification is accomplished through evaporation, precipitation with lanthanum hydroxide, and separation in an ion exchange column.

In the irradiated sulfur, the chemical form of the phosphorus is uncertain although it may be present as a mixture of elementary phosphorus and phosphorus sulfides (*3*). After melting, however, the P^{32} is almost certainly converted entirely to phosphorus pentasulfide which hydrolyses at the sulfur-nitric acid interface and passes into the aqueous phase as orthophosphoric acid. Aqueous phases other than dilute nitric acid are suitable for extracting P^{32} from molten sulfur, among them being strong nitric acid (*4*), 50% sulfuric acid (*5*), glacial acetic acid (*6*), and dilute sodium hydroxide (*7*). However, the dilute nitric acid extract is more easily purified than the others.

Extraction Vessel

The pressure vessel for the first stage of the extraction must be lined with material which will not contaminate the aqueous solution with metallic impurities or silica. The vessel must be safe against bursting and must also have provision for vigorous stirring of the contents, as the extraction entails reaction at the liquid-liquid interface.

A cylindrical pot with a wedge-shaped indentation across the bottom provides highly efficient stirring when rocked some 30 degrees either side of the vertical about an axis parallel to the indentation and at an oscillation frequency of 15–30 cycles/min. The design of the vessel is shown in Fig. 1.

These pots are of suitable size for extracting 1.8 kg of molten sulfur. They are mold-blown in Pyrex glass,

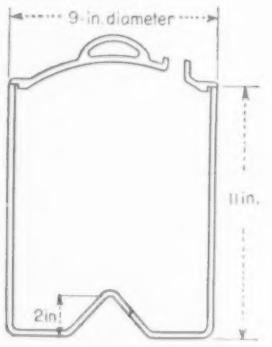

FIG. 1. Design of extraction vessel

although any other good borosilicate glass would be equally satisfactory. There is available commercially a Pyrex glass frying pan lid which was found to fit. At first, these lids were modified by attaching a short tube, as shown in Fig. 1, through which the aqueous phase of the contents could be withdrawn. This opening can be dispensed with, providing the operator is careful when removing the lid.

A stainless-steel container resembling a domestic pressure cooker (Fig. 2) keeps the inner pot under pressure at about 125° C during the extraction process. The glass pot is held in place with adjustable strips of steam packing material. The pressure container is gas heated and sealed by a rubber gasket. The operating pressure is 25–30 psi. The vessel rocks about an axis *A*. On the lid are fitted a large stainless-steel diaphragm-type pressure gauge and two stainless-steel safety valves. One of these is set to blow at 32 psi and the other at 35 psi. In addition, a needle valve is fitted for blowing off air when the vessel is heating up and also for equalizing pressures after cooling down.

Extraction Procedure

The lid of the outer vessel is removed and the water level checked. Two or three liters should be present. The lid of the inner pot is removed and, as it is probably active on the inside, it is laid on a sheet of cellophane active side downward. The lid is sufficiently thick to absorb all beta particles from P^{32} so that, until it is removed, no exposure of the operator is possible.

With the aid of tongs, a device *A*, shown in Fig. 3(a), is placed in the left-

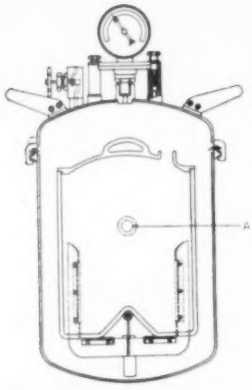

FIG. 2. Stainless-steel pressure heater containing extraction vessel

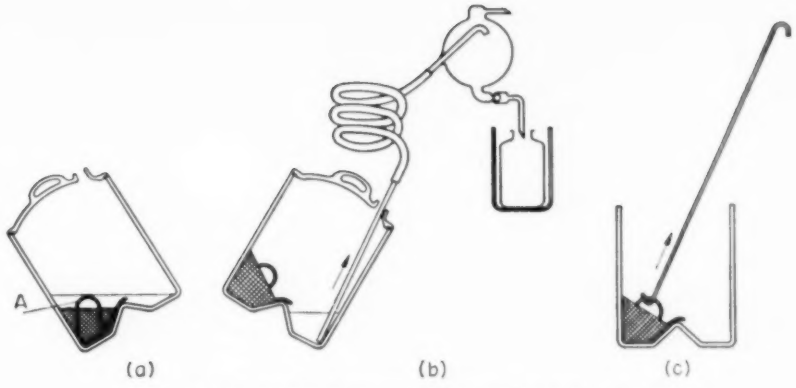

FIG. 3. (a) Initial placement of bent glass rod. (b) Removing aqueous phase after extraction. (c) Lifting out sulfur block

hand side of the pot. This is a stout glass rod bent into a form enabling it to sit in the inner pot and which, at the end of the extraction when the sulfur has resolidified, leaves two rings projecting from the block [Fig. 3(c)]. Six 300-gm blocks of irradiated sulfur are freed from their wrappings and deposited in the right-hand side of the pot with cutters and tongs operated through Perspex screening. This is followed by 1,500 ml of 0.2*N* nitric acid made up with very pure distilled water, and the inner lid is replaced.

The lid of the pressure vessel is put on and, with the needle valve open, heating is continued until all air is expelled and steam is issuing. The needle valve is closed and pressure raised to 30 psi. After allowing about half an hour for the sulfur to start to melt, the rocking mechanism is started.

After 3 hr, the gas heating is turned off and the rocking mechanism stopped with the vessel in the position of extreme inclination to the left as shown in Fig. 3(a). Almost all the sulfur is to the left of the indentation and forms a solid block about the glass framework. The vessel is then rocked over to the extreme right inclination [Fig. 3(b)], and the outer lid is removed. It is possible now to remove the aqueous phase through a glass tube and Polythene spiral by means of a vacuum hoist and deposit it into a shielded glass jar for transport to the purification apparatus. The sulfur block and inner pot are washed with a little more 0.2*N* nitric acid and the washings added to the main extract. If no second extraction is to be carried out on the batch of sulfur, the pot is turned to the vertical position and the inner lid removed. Long stainless-steel hooks are engaged in the glass rings projecting from the block, and in a few seconds it can be lifted out into a shielded container and sent for disposal.

The percentage extraction is measured by taking a few milligrams each of irradiated and extracted sulfur, dissolving in strong nitric acid, and comparing the activities due to P^{32}. With a 3-hr extraction, the yield at this stage is better than 70%.

Purification

Phosphorus in the crude extract may contain some sulfur as sulfate ion which will have S^{35} in it. There will certainly be metallic impurities and some organic matter. If silica should be present, it will not be removed by the purification process. However, since the Pyrex-glass-lined pressure extractor has been in use, it is very rare indeed for any silica to appear.

The four purification steps are:

1. Evaporation to about 80–100 ml.

2. Precipitation of lanthanum hydroxide in the solution, which holds the phosphate while any sulfate (carrying S^{35} activity) is filtered off to effluent.

3. Dissolving the lanthanum precipitate in hydrochloric acid, evaporation to dryness, and taking up in very dilute hydrochloric acid for purification in a cation-exchange-resin column.

4. Evaporation to dryness, destruction of the last traces of organic matter with 100-volume hydrogen peroxide and making up a stock solution.

Evaporation. In the apparatus shown in Fig. 4, which is screened by half-inch Perspex, stage **1** of the purification process is carried out in *A* to *G*. *A* is a parking place for a filter tube, *B* (sintered disk No. 0), which can be moved about on a Polythene spiral, *C*. The crude extract is lifted into the filter vessel, *D*, by the application of suction controlled by a stopcock. The vessel is fitted with a glass-ball nonreturn valve, *E*. The solution filters through a No. 1 sintered disk into the large evaporator, *G*. *G*, designed for fast evaporation from about 1,500 to 80 ml, consists of a vacuum-jacketed 3-liter flask of borosilicate glass; the liquid is heated by a 400-watt heater, *H*, wound round the protuberance on the bottom. Providing the liquid level is above the heater, local overheating does not take place, and there is little risk of pyrophosphate formation. Refluxing is avoided by the use of a pear-shaped stillhead. Unwanted water and acid distil at the rate of 5 ml/min with negligible loss of P^{32} activity by entrainment.

Lanthanum hydroxide precipitation. Stage **2** is carried out in a precipitator, *J* in Fig. 4, of similar type to those developed by Stang (*8*). A spherical joint has been adapted as a nonsticking ungreased stopcock. The socket, which acts as a delivery tube, is held against the ball by an adjustable counter-weight, the vertical thrust being applied effectively at the center of the ball. With the ball joint *M* turned until the hole in the ball is completely blanked off, clean compressed air is passed into the precipitator *J* through *L*. The solution from *G* is hoisted by suction into *I* and allowed to flow into *J*. It is held above the sintered disk (porosity No. 2) by the air stream. About 20 mg of lanthanum nitrate are washed in at *K* through a

curved funnel projecting through the Perspex screening. The lanthanum is then precipitated with ammonia which is also washed in. The air stream provides vigorous stirring. When the air is cut off, *M* is directed down a funnel *N* to the effluent reservoir. All the P^{32} is retained on the precipitate, but any S^{35} passes to effluent as ammonium sulphate. The precipitate is dissolved in concentrated hydrochloric acid, reprecipitated with ammonia, and refiltered.

Dissolving lanthanum precipitate. Stage **3** takes place in a small evaporator shown at *P* (Fig. 4). To avoid overheating even when dryness is reached, the inner evaporator vessel is heated by xylene vapor to 140° C. The distillation rate in this evaporator is better than 1 ml/min, and activity losses due to entrainment are negligible. When filtration in the precipitator is complete, the ball-joint *M* is closed and strong hydrochloric acid poured in through *K*. This is agitated vigorously with compressed air, and then *M* is directed to deliver solution into *P*. The precipitator is washed with more acid and then water, all the washings being added to the contents of *P*, and evaporated to near dryness.

Some batches of target sulfur have a rather high arsenic content which occasionally appears as arsenate ion in the final product. Its removal can be carried out at this stage in *P* by reducing with hydrobromic acid to arsenic tribromide and distilling off the tribromide, excess hydrobromic acid being destroyed with 100-volume hydrogen peroxide (*9*).

After evaporation to dryness, a few drops of 100-volume hydrogen peroxide are added to destroy organic matter, and, after reaction, more is added. Evaporation is then again carried through to near dryness.

Practically all cationic impurities, including the lanthanum added earlier, are removed in an Amberlite IR-100 column 14 mm in diameter and 15 cm in length; little or no phosphate ion remains on the column. This step is best carried out at an acidity of 0.05*N* (*9*). To the contents of *P* (Fig. 4) are added through *O*, 25 ml of 0.1*N* hydrochloric acid; this is boiled to bring all the phosphate into solution. 25 ml of pure distilled water are added and the mixture, now approximately 0.05*N* with respect to hydrochloric acid, is raised in a suction hoist, *Q*, and allowed to flow through the column *R* into a final evaporator, *S*. When the level of the solution is below the bulb of *R*, a further 50 ml of 0.04*N* hydrochloric acid is added to *P*, hoisted in *Q*, and allowed to flow into *R*.

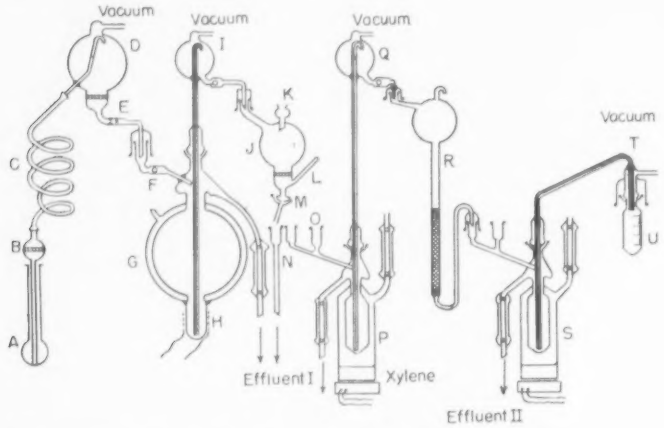

FIG. 4. Apparatus for purifying P^{32} obtained in extraction process

Evaporation and storage. While the second 50 ml of acid is going through the column, the first 50 ml is being evaporated in *S*. After evaporation to near dryness, last traces of organic matter are destroyed with two small additions of 100-volume hydrogen peroxide of microanalytical reagent quality followed by evaporation to near dryness with one lot of double-distilled, pyrogen-free water. The P^{32} is finally taken up in about 50 ml of double-distilled pyrogen-free water. It is usually at *p*H 2–3.

The storage bottle, *U*, is fitted with a spherical ground stopper and dust cap. Transfer of stock solution to the storage bottle in its Perspex shield is carried out by suction at point *T*. When transfer is complete, the vacuum is automatically broken and the storage bottle is removed and stoppered. After spectrographic analysis and radioactive standardization, the stock solution is ready for dispensing.

Cleaning out the apparatus after a preparation is straightforward. If *B*, *C*, *D* and *G* (Fig. 4) need cleaning, which is not usually the case, 2*N* hydrocholoric acid can be sucked in and passed through *I*, *J* and *N* to effluent, followed by pure water. The column *R* is cleaned by passing first 6*N* hydrochloric acid and then plenty of pure water through *O*, *P*, *Q* and *R* and thence to *S*. An effluent flask shaped like a storage bottle can then be attached at *T* for removing washings.

All shielding of the apparatus is by half-inch-thick Perspex. Ports are provided at the effluent points *I* and *II* (Fig. 4) for effluent removal and at *U* for removal of finished solutions. Reagents are introduced through curved funnels emerging through the front of the Perspex shield. The only other controls are a small mechanism for operating the ball joint *N* (Fig. 4), four two-way stockcocks to control the solution hoists, the compressed-air control for the Stang precipitator, and the electrical controls for the evaporator heaters.

It is possible to process up to four lots of P^{32} at the same time—one being finished off at *P*, *R* or *S* (Fig. 4), one filtering in *J*, one evaporating in *G*, and a fourth extracting in the pressure vessel. The total time needed for a single extraction is about two days. If carrier is added to the original extract to the extent of about 20 μg/mc, very little P^{32} is lost, but preparation of material without added carrier gives a yield of about 60–70% in the purification process.

* * *

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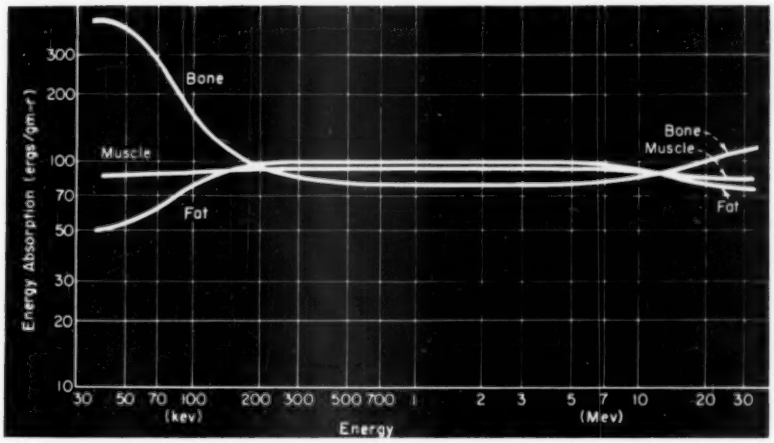

FIG. 1. Variation of energy absorption for different tissues

Production of Multicurie Gamma-Ray Teletherapy Sources

Both production problems and source characteristics must be taken into account in choosing a teletherapy isotope. Selection of most suitable source is discussed, and cost data and details of reactor production are given for 14 isotopes

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As NUCLEAR REACTORS with high neutron fluxes become more numerous, there is increasing interest in the use of reactor-produced isotopes as sources for deep therapy. A number of multicurie radioisotope sources are now in use and it is becoming evident that many more will be used in the future.

It is the purpose of this paper to discuss the selection of the most suitable isotope for a teletherapy source and to outline the problems to be expected in its production. Especially detailed consideration is given to the problem of neutron shielding within the target material, and experimental results concerned with evaluating the effects of shielding are presented.

Source Requirements

Certain requirements must be met by any source that is to be useful for treatment of deep-seated cancer. The main factors to be considered in choosing a source are energy of the γ-radiation, half-life, optimum treatment distance, roentgen output at this distance, and minimum source diameter and its attendant penumbra.

Energy of the radiation. The biological effects of any radiation result from the ionization produced on passage through tissue. Radiation sensitivity of cells appears to vary with their rate of growth and, for this reason, cancer cells are in general more easily destroyed than neighboring normal cells. However, this differential is not sufficient to allow broad-beam irradiation, and it is necessary to try to get maximum concentration of the ionization at the location of the cancer cells. Involved in this problem is the energy of the radiation used. At low energies, the maximum ionization occurs at the

skin, and as the energy increases, the maximum moves progressively deeper.

The secondary electrons, projected with high velocity by the incident radiation, are the prime ionizing agents, and the pattern of projection varies with the energy. At low energies, photoelectric interaction causes emission of electrons at right angles to the incident beam, whereas at higher energies, Compton scattering predominates and the probability of forward scatter is high. Thus in the photoelectric range there is more backscatter (which increases the skin dose) and more side-scatter out of the beam (which increases the total body dose). For the effective atomic number of tissue, the cross section for photoelectric capture rises rapidly below 400 kev, so that, from scattering considerations, the energy used should be above this minimum.

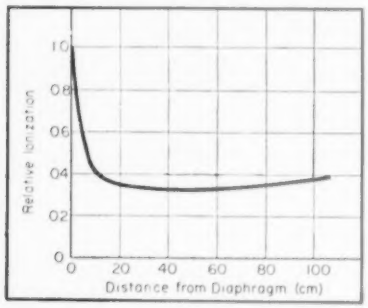

FIG. 2. Relative ionization at various distances from a defining aperture

Skin cells are constantly growing and therefore are radiation-sensitive, so that dosage at the skin should be kept down. The effective atomic number of skin cells is raised by a relatively high sulfur content, and for this reason their absorption in the photoelectric range is high. This is illustrated by experiments reported by Quimby (*1*) in which 270 r of 100-kv (25-kev) X-rays produced a skin erythema, whereas 1,000 r were needed at 1,000 kv (400 kev).

Bone contains radiation-sensitive blood-forming cells, and consequently should receive low dosage. Photoelectric absorption increases with the fourth power of atomic number, and since bone has an effective atomic number of 13.6, compared to 7.4 for muscle and 5.9 for subcutaneous fat, it is evident that photoelectric absorption in bone will be comparatively high. On the other hand, the cross section for pair production increases with the second power of the atomic number and predominates for high energies, so that bone absorption will increase again in the high-energy region. These effects are illustrated in Fig. 1, which is adapted from Spiers and Johns (*2, 3*).

Shielding requirements in the source container and in the treatment room increase with increasing energy. The half-value thickness of lead is 1.4 cm at 2 Mev, compared to 0.4 cm at 0.25 Mev, and since the weight of the protective housing varies as the cube of its linear dimensions, the expense of shielding rises rapidly as the energy goes up. On this count, energies should be kept down, but no practical difficulty has been experienced in designing shielding for therapeutically useful intensities of the 1.2-Mev energy of Co^{60} radiation.

There are definite biological advantages in the use of high energies. From scattering considerations and bone-damage considerations, the energy should be well into the Compton range (above 400 kev), and from depth-dose considerations, the higher the energy the better. Increased cost of shielding must be considered for higher energies, but is not prohibitive at 1.2 Mev.

Half-life. Convenience and economy govern the choice of half-life in a beam therapy source. Frequent correction in treatment times becomes necessary if the decay is too rapid. Thus, for a half-life of 6 months, the output drops about 5% every 2 weeks, necessitating adjustment of treatment time. For a 6-yr half-life, adjustment every 6 months is sufficient.

Replacement at least every half-life is necessary, and the expense and inconvenience of this operation, combined with troublesome dosage adjustment, point to a practical minimum half-life of about 3 months. Practical maximum half-life will be set by cost of activation.

Optimum treatment distance. Inverse-square-law dependence influences the optimum treatment distance. To illustrate this, the ratio of dose at a depth of 2 cm to surface dose may be compared for two typical cases, one a radium teletherapy unit with a 10-cm source-to-skin distance and the other an isotope unit with a 100-cm source-to-skin distance.

For the radium unit, neglecting absorption, the ratio will be $(10/12)^2 = 0.69$ and for the isotope unit it will be $(100/102)^2 = 0.96$.

It will be seen from this comparison that a considerable gain in depth dosage is obtained by increasing the source-to-skin distance. The optimum point is reached when the source-to-skin distance is large enough to give satisfactory depth dose without unnecessary sacrifice of source strength.

From the standpoint of depth dosage alone, treatment distances between 70 and 125 cm are desirable.

Roentgen output. The problem of determining the best possible dosage rate is a biological one. Much clinical data has been accumulated concerning different types of tumors and the most effective treatment for each. For present purposes it is sufficient to take 20 to 30 r/min as a satisfactory dosage rate from an isotope therapy unit.

A factor to be considered in this connection is that the number of patients that can be treated daily is determined by the roentgen output, and dosage rates below 20 r/min necessitate longer treatment times, which puts a limit on this number.

Source diameter. A well-defined beam serves to localize the tissue damage and keep the whole-body dose to a minimum. Good definition depends on the size of the source and its distance from the defining aperture and from the skin. Since a circular source is most convenient to fabricate and handle, the important dimension of the source will be its diameter.

Distance from the defining aperture or diaphragm to the skin is governed by considerations of skin dose. High-speed secondary electrons are ejected from the defining material and travel a considerable distance in air. Since these would all be absorbed by a few millimeters of tissue at the surface, unduly increasing the skin dose, it is necessary to make the diaphragm-to-skin distance large enough to minimize the effect of these secondary electrons.

Figure 2 shows a plot of relative ionization at different distances from the diaphragm. The curve was adapted from measurements on a 2-Mev X-ray generator by Howarth *et al.* (*4*) and on a Co^{60} therapy unit by Johns *et al.* (*5*). It will be seen from this curve that a distance of 20 cm between diaphragm and skin is satisfactory.

Using this distance, the penumbra associated with a source may be deter-

mined by simple laws of geometrical optics as

$$x = D[20/(a - 20)] \qquad (1)$$

where x is penumbra width in cm, D is source diameter in cm, and a is treatment distance in cm.

Penumbra widths up to 1.25 cm are acceptable on biological grounds, and on this basis the maximum permissible source diameter is 5 cm for treatment distance of 100 cm. This value has been used in the calculations to follow.

Pile Production Problems

In practice it is only possible to select an isotope with characteristics most nearly approximating those desired. A cursory examination of known isotopes will show several which meet the requirements of gamma energy and lifetime. However, the only potentially useful ones are those which can be produced with specific and total activities great enough to give the output desired with the diameter selected. A series of calculations is necessary for each isotope, and the possible ones may then be compared with respect to cost of production.

Specific activity. Consideration of the self-absorption of its own radiation by the source shows that the minimum specific activity necessary for a certain output is governed by the source diameter. Dixon (*6*) has treated the problem of self-absorption for various source shapes. For the case of a cylindrical source, measured at large distances on the longitudinal axis, the ratio of intensity, I, with self-absorption to that, I_0, with no self-absorption is

$$\frac{I}{I_0} = \frac{1 - e^{-\mu r}}{\mu r} \qquad (2)$$

where μ is the linear absorption coefficient of the material for its own γ-rays, and r is the length of the source along the axis of measurement. As r increases, I tends to a value I_{max} such that

$$I_{max} = C\rho S/\mu \qquad (3)$$

where C is the cross sectional area, ρ is the density, and S is the specific activity.

Figure 3 shows percentage of intensity transmitted as a function of source length for different absorption coefficients, and Fig. 4 shows percentage of maximum possible output obtained for different lengths of source for a number of materials having different absorption coefficients. These graphs show that increasing the quantity of active material soon adds nothing to the roentgen output if the source diameter remains the same. Higher output can only be obtained by increasing the specific activity.

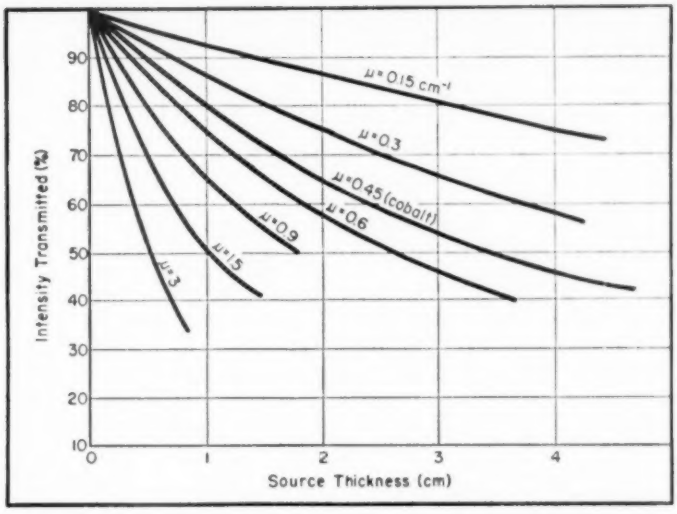

FIG. 3. Variation of transmitted intensity with source thickness for a number of materials

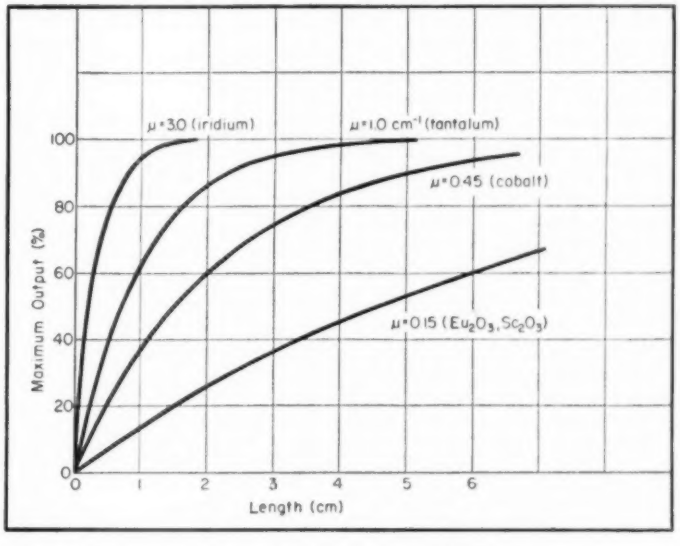

FIG. 4. Maximum output as a function of source thickness

The minimum specific activity needed to give a certain output may be determined from Eq. 3 as

$$S_{min} = \mu I/C\rho \qquad (4)$$

This minimum specific activity may not be possible for a given isotope if the neutron flux available is not high enough.

The activation equation is

$$S = \frac{\phi N\sigma}{W(3.7 \times 10^{10})}(1 - e^{-0.693t/T}) \qquad (5)$$

where ϕ is neutron flux, N is Avogadro's number, σ is activation cross section, W is atomic weight, t is irradiation time, and T is half-life, in the same units as t. Saturation is reached when t/T becomes large, in which case the maximum specific activity obtainable is expressed by

$$S = \frac{\phi N\sigma}{W(3.7 \times 10^{10})} \qquad (6)$$

Comparison of values given by Eqs. 4

and 6 will show if a source that is to give a certain output can be prepared with a given neutron flux.

Irradiation time. Growth of specific activity follows the exponential part of Eq. 5. Figure 5 illustrates this growth and shows that it becomes uneconomical to use long irradiation times. Lengthening irradiation time by 100%, from one half-life to two half-lives, will only increase the activity by 50%.

On the other hand, the comparative inefficiency of using long irradiation times and high flux must be balanced against the relative cost of different flux positions. Charges for irradiation are lower in low flux positions since reactor operation is less affected by absorptive material in these positions.

Irradiation efficiency. Due to the shielding effect of the target material, the neutron flux passing any point in the target will be less than it would be in the absence of a target. On this account, the actual activity produced will be less than that calculated from Eq. 5. The ratio of the activity produced to the activity if there were no shielding is called the irradiation efficiency, and it is necessary to consider this quantity in calculations of activities.

For plane targets, a method due to Kennedy (7) may be applied. Let I_0 be the intensity of the neutron flux per unit solid angle. The number of neutrons incident on 1 cm² of area between θ and $\theta + d\theta$ will be $2\pi I_0 \sin\theta \cos\theta\, d\theta$, where θ is the angle to the normal. The intensity transmitted through a layer of thickness x will be

$$I_2 = \int_0^{\pi/2} I_0 2\pi \sin\theta \cos\theta\, e^{\frac{-\Sigma x}{\cos\theta}} d\theta \quad (7)$$

and

$$\Sigma = \frac{N\sigma_a\rho}{W} \quad (8)$$

where Σ is the linear absorption coefficient for thermal neutrons, σ_a is the thermal-neutron absorption cross section, ρ is the density, W is atomic weight, and N is Avogadro's number. The fraction of incident neutrons transmitted is

$$F(x) = \frac{I_2}{2\pi \int_0^{\pi/2} I_0 \sin\theta \cos\theta\, d\theta} = \frac{I_2}{\pi I_0}$$

$$= 2\int_0^{\pi/2} \sin\theta \cos\theta\, e^{\frac{-\Sigma x}{\cos\theta}} d\theta$$

The substitution $1/y = \cos\theta$ gives

$$F(x) = 2\int_1^{\infty} \frac{e^{-\Sigma x y}}{y^3}\, dy \quad (9)$$

which may be integrated for different values of x with the curve of $\int_a^{\infty} \frac{e^{ay}}{y}\, dy$ given by Jahnke and Emde (8).

The most practical method of irradiating targets, in the case of the Chalk River NRX reactor, is in the form of cylindrical shells since the standard irradiation capsules are cylindrical in shape. Rigorous calculation of the shielding in this case is difficult and it is necessary to use an approximation. The cylindrical shell may be considered as flattened into a rectangular plate with thickness equal to twice the wall thickness of the shell.

To establish the validity of this approximation, experiments have been carried out, and the results are presented in Table 1 and Figs. 6 and 7. Four targets were prepared of cobalt pellets 1 mm in diameter, 1 mm long, and with an effective density of 4.77 gm, arranged in the form of cylindrical shells of length 4.05 cm, outside diameter 1.86 cm, and wall thickness approximately 1, 3, 5, and 7 mm, respectively. These targets were irradiated in accurately comparable neutron fluxes for a total of approximately 1.73×10^{19} *nvt* each.

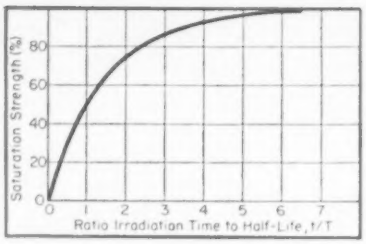

FIG. 5. Saturation strength as a function of irradiation time

Absolute value of the flux does not enter into the experiment, and relative flux for each position was accurately determinable. Roentgen outputs from the sources were measured on an axis at right angles to the longitudinal axis of each cylinder. Corrections to be applied for self-absorption in the shells were determined from Fig. 6.

In this figure, the calculated value of the self-absorption of a solid cylinder is plotted as point A, using the method of Dixon (6). For very small wall thickness, the self-absorption will approach zero, and for other thicknesses between zero and the maximum, when the wall thickness becomes equal to the radius of the cylinder, the self-absorption will have a magnitude intermediate between the extremes.

The dotted curve in Fig. 6 joins approximate values calculated on the assumption that the cylindrical shell is flattened, in the way described above, into a rectangular plate with thickness equal to twice the wall thickness of the shell. This curve demonstrates that straight-line interpolation is justified to obtain the solid curve of the figure. In this way, transmission factors are obtained for the wall thicknesses used in the experiment.

In Fig. 7, the curve marked $F(x)$ represents the results of integration of

TABLE 1—Results of Irradiations of Cylindrical Shells of Cobalt Pellets

Shell-wall thickness (mm)	*Mass of cobalt pellets (gm)*	*Relative total nvt received*	*Output corrected for nvt received (rhm)*	*Transmission factor from Fig. 6*	*Measured output (corrected) (rhm)*	*Output if no neutron shielding* (rhm)*	*Irradiation efficiency* (%)*
1.03	10.7	1.000	2.98	0.982	3.03	3.56	85
3.06	28.5	1.012	4.62	0.945	4.89	8.06	60.7
5.10	41.4	1.018	5.14	0.909	5.66	12.5	45.3
7.14	52.3	0.994	5.72	0.872	6.56	14.8	44.3

* 85% efficiency assumed for source No. 1.

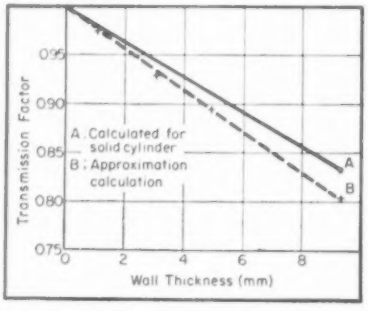

FIG. 6. Determination of self-absorption corrections for trial sources

Eq. 9 for a number of values of x. The irradiation efficiency may be determined from this by finding the area under the curve up to a particular value of x and dividing this area by the value of x. Repetition of this process yields the curve of calculated irradiation efficiency. The thermal-neutron absorption cross section used in this calculation was 34.2 barns, following Yaffe *et al.* (9).

Table 1 lists the data for the experiments. The figures for output if there is no shielding were determined by assuming an irradiation efficiency of 85% for source No. 1. This efficiency was arrived at by successive approximations, taking account of the fact that the efficiency will reach 100% for wall thicknesses approaching zero. Comparison of these figures with the measured outputs yields the irradiation efficiencies, which are also plotted in the dotted curve of Fig. 7. Vertical bars through the points of this curve represent estimated outside limits of error arising from probable errors of ±5% in the measured outputs.

The calculated irradiation-efficiency curve of Fig. 7 may be used to determine efficiencies to be expected for other target materials by noting that for a value of Σ which differs from that for cobalt, a corresponding value for x may be found such that the product Σx is the same. For example for cobalt, Eq. 8 gives $\Sigma = 1.68$, and for iridium, using a value of $\sigma_a = 472$ barns and $\rho = 16$ gm/cm³, the value is $\Sigma = 23.7$. Dividing the abscissas of Fig. 7 in the ratio of $23.7/1.68 = 14.1$ determines the efficiency curve for iridium. For example, at 70% efficiency Fig. 7 gives $x = 2.5$ mm for cobalt, or $2.5/14.1 = 0.18$ mm for iridium, indicating that an irradiation efficiency of 70% may be expected for cylindrical shells of iridium metal with a wall thickness of 0.18 mm.

Cost factor. When the roentgen output desired has been established, the number of curies necessary to produce it can be calculated. For example, in each disintegration of Co^{60} two gamma rays are emitted in cascade, with energies of 1.17 and 1.33 Mev. Since the linear absorption coefficients in air for these energies are 3.53×10^{-5} and 3.41×10^{-5} per cm, respectively, the absorption in 1 cm of air is

$$(1.17 \times 3.53 \times 10^{-5} + 1.33 \times 3.41 \times 10^{-5}) = 8.68 \times 10^{-5} \text{ Mev}$$

A roentgen is, by definition, the amount of radiation that will produce one electrostatic unit of charge in one cubic centimeter of air at normal temperature and pressure. It has been determined that 32.5 ev are expended to form one ion pair. Since 1 r produces 2.083×10^9 ion pairs, it corresponds to the absorption of 6.77 $\times 10^4$ Mev/cm³ of air. Hence, 1 curie of Co^{60} will produce

$$\frac{3.7 \times 10^{10} \times 3{,}600 \times 8.68 \times 10^{-5}}{4\pi \times (100)^2 \times 6.77 \times 10^4} = 1.3 \text{ rhm}$$

where rhm means roentgens per hour at 1 meter.

This calculation may be carried through for any other case when the disintegration scheme is known.

The specific activity required is found from

$$S = \frac{RK}{G} \frac{(100 + E)}{100} \qquad (10)$$

where S is specific activity in curies/gm, R is roentgen output in rhm, K is the number of curies to give 1 rhm, E is per cent self-absorption in the source, and G is mass of the source in grams. Using Eq. 5, the irradiation time to give this specific activity may be found, a correction being applied for the irradiation efficiency.

Cost of irradiation may be expressed as

$$\text{Cost} = \frac{AG}{U} \phi t P \qquad (11)$$

where A is total thermal-neutron absorption cross section in cm²/gm, G is total mass, in gm, of target material in an irradiation container, U is maximum cross section, in cm², allowed per container, ϕ is thermal-neutron flux in n/cm²/sec, t is irradiation time in sec, and P is the price of 1 neutron per cm² (set by economics of reactor operation). If AG/U is fractional, the next largest integer is used, and if more than one irradiation container is required, the effective value of AG/U will be the number of containers necessary to hold the material.

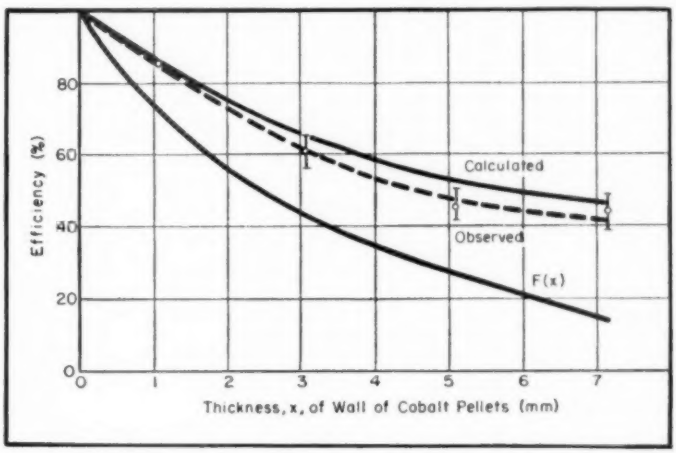

FIG. 7. Comparison of calculated and observed irradiation efficiencies

In some cases, the cost of the target material will be significant and it will be necessary to append this cost to the figure, obtained from Eq. 11.

Comparison of Isotopes

A compilation of isotopes suitable for multicurie gamma sources is given in Table 2. A source strength of 20 r/min at 1 meter is used for the calculations and 25% self-absorption is assumed. The irradiation fluxes are those commonly used at the Chalk River NRX Reactor.

The column listing number of irradiation containers gives the minimum number of standard irradiation capsules (cylinders, height 4.05 cm, inside diameter 1.9 cm) required to hold the target material. This is the value of AG/U used in Eq. 11 to calculate the irradiation cost. Values of U, the maximum absorption cross section allowed per capsule used in these calculations are 2.12, 4.00, 6.40, 12.5, and 34.5 cm² for the flux positions 5.7,

TABLE 2—Comparison of Various Isotopes Considered as Possible Teletherapy Sources

Isotope	Target material and density (gm/cm³)	Half-life	Gamma energy (Mev)	Cross sections (barns)		$\Sigma = \frac{N\sigma_a\rho}{W}$	rhm per curie	Curies for 20 rmm (25% self-absorption)	Height of source (cm) (25% self-absorption)	Source diameter (cm)	Mass of source (gm)	Specific activity of isotope (curies/gm)	Fluxes ($\times 10^{13}$ n/cm²/sec)		Number of irradiation containers	Wall thickness (mm)	Irradiation efficiency (%)	Irradiation time (days)	Relative irradiation cost
				Activation	Total absolute								Minimum	Irradiation					
Ag^{110}	Metal 10.5	270 d	0.6, 0.9 1.4, 1.5	1	60	0.35	1.4	1,130	1	5	210	5.4	4.5	—	—	—	—	—	—
Ba^{133}	Ba_2CO_4 2	>20 y	0.1, 0.32 0.36	0.006	1.4	0.012	0.25	6,000	3	5	120	76	10^3	—	—	—	—	—	—
Co^{60}	Metal pellets 4.77	6.26 y	1.17 1.33	34	34	1.68	1.3	1,230	2.43	2	36.3	33.9	0.36	5.7	6	0.63	90	206	37.2
														4.3	4	1.0	85	271	24.1
														3.4	2	3.8	76	414	14.6
														0.7	2	3.8	76	2,970	20.6
										3	81.9	15.0	0.16	5.7	14	0.63	90	92	37.5
														4.3	8	1.1	84	128	23.0
														3.4	5	1.8	77	177	15.6
														0.7	4	2.1	74	995	10.8
										4	145	8.48	0.09	5.7	24	0.63	90	52	37.5
														4.3	13	1.2	83	71	20.7
														3.4	8	1.8	77	99	13.6
														2.0	6	2.6	70	183	11.4
														0.7	6	2.6	70	530	11.5
										5	227	5.40	0.06	5.7	38	0.63	90	33	36.6
														4.3	20	2.5	82	46	20.5
														3.4	13	2.0	75	62	14.4
														2.0	9	2.6	70	109	10.2
														0.7	9	2.6	70	330	10.8
Cs^{134}	Cs_2CO_3 powder 1.9	2.3 y	0.56 0.60 0.79	26	36	0.25	1.15	1,390	4	2	23.9	71.4	2.2	5.7	2	3.4	92	676	39.6
														4.3	1	9.5	80	—	—
														3.4	1	9.5	80	—	—
														0.7	—	—	—	—	—
										3	53.8	31.9	1.0	5.7	4	4.0	91	263	31.5
														4.3	3	6.3	86	375	25.4
														3.4	3	6.3	86	508	26.9
										4	95.4	17.9	0.6	5.7	7	4.0	91	147	30.3
														4.3	5	7.0	84	208	23.2
														3.4	5	7.0	84	250	22.0
										5	149	11.5	0.4	5.7	11	4.0	91	89.0	29.0
														4.3	7	9.5	80	135	21.1
														3.4	7	9.5	80	165	20.2
														0.7	7	9.5	80	1,120	28.5
Eu^{152} Eu^{154}	Eu_2O_3 2.3	16 y (10, 11)	0.122, 0.123 0.244, 0.344 0.720, 0.778 0.964, 1.08 1.11	1,325 (1,200 + 125)	4,200	33	0.55	2,900	4	1	7.22	465	3.3	5.7	58	0.03	91	540	925
														4.3	31	0.05	85	723	496
														3.4	20	0.08	79	1,090	383
										2	29.0	116	0.8	5.7	232	0.03	91	137	940
														4.3	124	0.05	85	182	500
														3.4	78	0.08	79	253	347
														0.7	47	0.13	71	1,440	249
										3	65.2	51.5	0.4	5.7	525	0.03	91	60.5	938
														4.3	280	0.05	85	84.8	525
														3.4	174	0.08	79	115	352
														0.7	105	0.13	71	618	235

Ir^{192}	Metal powder 16	74 d	0.21 0.30 0.47 0.60	285	472	23.5	0.27	6,000	0.2	2	10.1	595	0.02	5.7 4.3 3.4	7 4 3	0.05 0.06 0.10	91 87 81	66.7 92.0 163	13.8 8.64 8.58
										3	22.5	267	0.01	5.7 4.3 3.4	16 9 6	0.05 0.06 0.10	91 87 81	26.4 37.7 52	12.3 7.56 5.46
										4	40.3	149	0.006	5.7 4.3 3.4 0.7	28 15 10 7	0.05 0.08 0.12 0.17	91 83 78 72	14.3 20.2 27.7 305	11.9 6.75 4.86 7.74
										5	62.8	95.5	0.004	5.7 4.3 3.4 2.0 0.7	44 23 15 10 10	0.05 0.08 0.12 0.19 0.19	91 83 78 70 70	8.7 12.7 16.7 34.3 130	11.3 6.51 4.38 3.55 4.61
Sb^{124}	Metal 6.6	60 d	0.1, 0.6 1.7, 2	1.1	8.3	0.27	1.2	1,330	1	5	130	10.3	0.7	—	—	—	—	—	—
Sb^{125}	Tin 7.2	2.7 y	0.2, 0.4 0.6	0.4	0.7	0.02	0.72	2,230	1	5	145	15.4	280	—	—	—	—	—	—
Sc^{46}	Sc_2O_3 2	85 d	0.18 0.89 1.12 2	22	32	0.85	1.1	1,470	4	2	23.9	93.4	1.2	5.7 4.3 3.4	5 3 2	1.2 2.0 3.2	91 85 78	32.7 46.0 65.5	4.80 3.06 2.31
										3	53.8	41.8	0.5	5.7 4.3 3.4 0.7	11 6 4 3	1.2 2.2 3.7 5.3	91 84 76 70	14.0 19.2 27.6 240	4.56 2.55 1.95 2.61
										4	95.4	23.6	0.3	5.7 4.3 3.4 0.7	20 10 7 6	1.2 2.3 3.7 4.7	91 83 76 72	7.8 11 15 95	4.62 2.46 1.86 2.07
										5	149	15.1	0.2	5.7 4.3 3.4 2.0 0.7	30 16 11 9 9	1.2 2.3 3.7 4.9 4.9	91 83 76 71 71	5.1 6.6 9.2 17 54	4.50 2.40 1.80 1.60 1.77
Sm^{145}	Sm_2O_3 2	150 d	0.24 0.95	?	8,100	98	0.68	2,360	3	5	120	23.0	?	—	—	—	—	—	—
Se^{75}	Metal 4.3	127 d	0.1, 0.3 0.4	0.2	12.5	0.4	0.30	5,330	1.5	5	130	41.0	98	—	—	—	—	—	—
Ta^{182}	Metal 16.6	115 d	1.13 1.22	20.6	21	1.16	0.61	2,670	0.6	2 3	31.2 70.3	85.7 38.0	5.0 2.0	5.7 5.7 4.3 3.4	1 3 2 1	0.9 0.7 1.1 2.1	90 92 89 80	270 82 110 194	8.10 7.53 4.89 3.39
										4	124	21.5	1.1	5.7 4.3 3.4	5 3 2	0.7 1.2 1.8	92 87 82	41.2 60.0 83.0	6.15 4.02 2.91
										5	494	13.7	0.7	5.7 4.3 3.4	7 4 3	0.8 1.4 1.9	91 85 81	25.3 37.0 49.5	5.22 3.30 2.61
Zn^{65}	Metal 7.1	250 d	1.14	0.25	1	0.06	0.30	5,330	2	5	280	19.0	30.2	—	—	—	—	—	—
Zr^{95}	ZrO_2 2	65 d	0.73 0.23	0.01	0.4	0.005	0.40	4,000	4	5	160	38.0	10^4	—	—	—	—	—	—

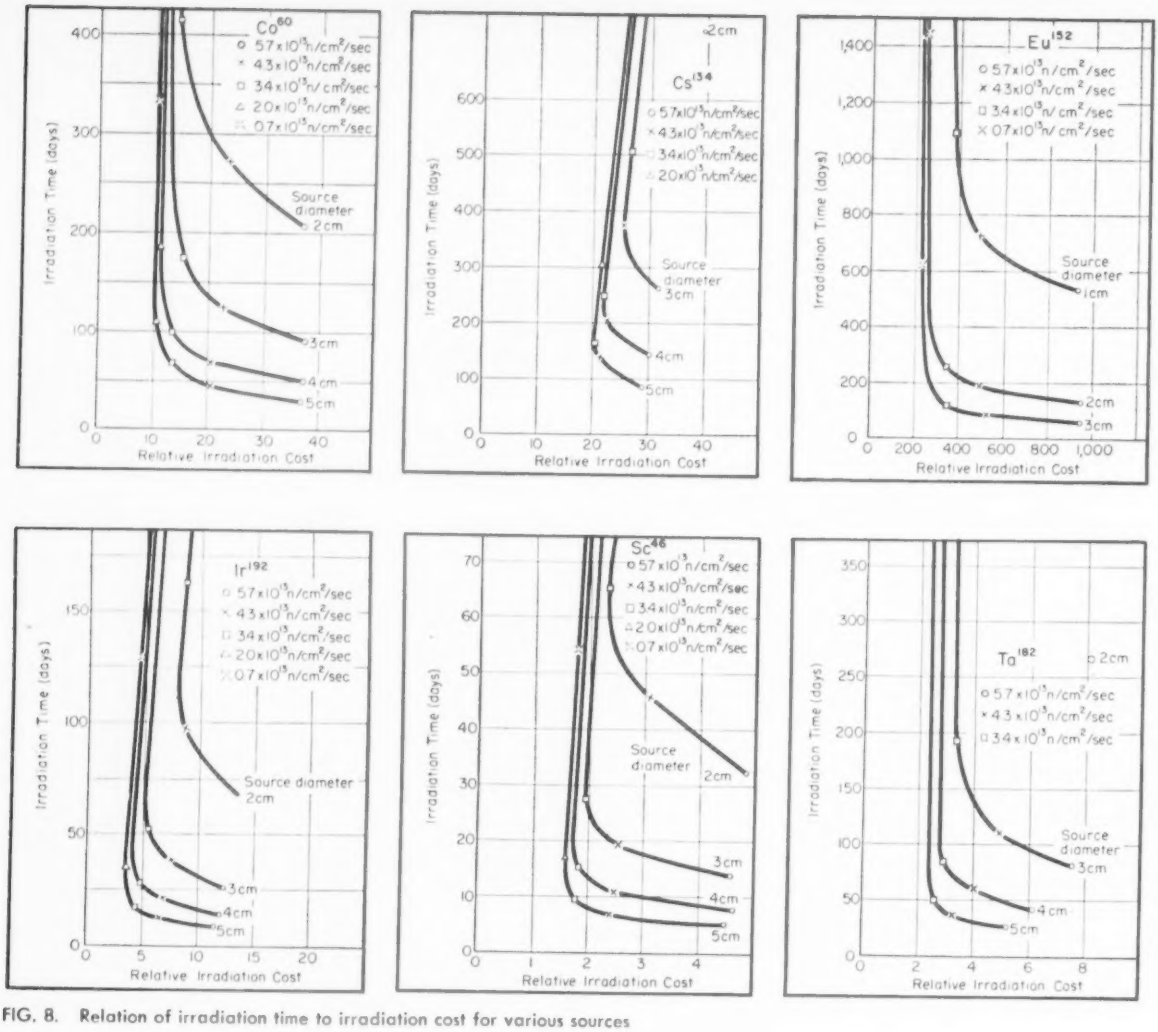

FIG. 8. Relation of irradiation time to irradiation cost for various sources

4.3, 3.4, 2.0, and 0.7 × 10^{13} n/cm²/sec, respectively.

In some cases the number of containers is set by the physical volume which can be put in the container, and in other cases it is set by the maximum wall thickness which will permit at least 70% irradiation efficiency. The column of irradiation efficiencies lists values calculated in each case for the appropriate wall thickness, using the value of Σ and the curve of Fig. 7.

Graphs of irradiation time vs irradiation cost are plotted in Fig. 8 for each of the isotopes in Table 2 which might be used for a multicurie therapy source.

In Table 3 are listed the costs, over a 10-yr period, for therapy sources using the isotopes of this group. The number of replacements is determined on the assumption that a replacement is required every half-life. The flux position listed in each case is determined from the appropriate graph in Fig. 8 to give the lowest cost, with reasonable irradiation time. The total cost involves the price of target material required for one source, the initial irradiation price, and the price of the re-irradiations required during the 10-yr period. Exchange costs are estimates of the price of the handling and shipping involved during an exchange. This price, and the material cost, can only be estimated approximately, but large discrepancies here will not alter the ordering of the final totals.

Fission-Product Sources

At present, the use of fission products as teletherapy sources does not seem practicable. Gross fission products are ruled out by low specific activity and complex decay rate. A detailed discussion of the difficulties of extraction of active material suitable for therapy sources from fission products has been given recently by Rupp (*12*).

Only Ce^{144} and Cs^{137} offer promise of usefulness, and problems of separation and concentration into the high-specific-activity sources necessary for therapy have not been overcome. The particular case of Cs^{137} has been considered by Eastwood (*13*). In this connection, the low output per curie must be taken into account. This is 0.34 rhm/curie as compared to 1.3 rhm/curie for Co^{60}, so that the source strength required for a given roentgen output will be three or four times greater than that of Co^{60}.

Handling and Shielding

The high-activity sources required in teletherapy units give rise to handling and shielding difficulties on a scale not ordinarily encountered. Lead and tungsten shielding has been employed in the treatment heads of existing units, and shutter mechanisms based on pumping mercury in and out of a gate and rotating the source from an "off" to an "on" position have been used (*14–16*).

The general problem of room protection has been discussed by Dixon, Garrett, and Morrison (*17*) for both direct beam shielding and scattered radiation shielding.

Shipping, installing, and exchanging the sources require arrangements for continual protection of the operators. A technique has been evolved (*18*) which utilizes a matching transfer case and treatment head. The source container is mounted in a transfer-case drawer for shipment. Installation is accomplished by sliding the drawer from the transfer case into the treatment head.

Since efficient irradiation requires disposal of the material in thin sections within the irradiation container, the problem of transfer into a final source container arises. In view of this, the physical form of any source material must be carefully considered. For the early cobalt units, metallic disks and wafers were used, but recent practice has been to use small cylindrical pellets 1 mm in diameter and 1 mm long. A technique has been devised for transferring these from irradiation containers, in which they are arranged in cylindrical shells, into final source containers. For any other isotope, a corresponding technique would be required.

Conclusions

Costs of a deep therapy source using Co^{60} are about half those for alternative pile-produced isotopes. Sc^{46} and Ta^{182} are next on an ascending price scale, but both have the additional disadvantage of short half-life.

There has been considerable interest in the possibility of using europium because of its high activation cross section which would appear to make possible high specific activity with low neutron flux irradiation. However, comparative costs make it unlikely that europium sources can become competitive with cobalt. The high neutron absorption coefficient means that only extremely thin sections can be irradiated for reasonable efficiencies and for permissible pile loading. This necessitates distributing the target material through a large volume in the reactor and, since irradiation price necessarily depends on volume occupied, the advantage of high activation cross section is outweighed. In addition, as little as 30% of the neutron flux available may be contributing to the activation because of the unfavorable ratio of activation to absorption cross section. Comparison of the figures in Table 3 shows that the total costs for europium are much higher than for cobalt although allowance is made for longer half-life, with consequent reduction in the number of source replacements necessary.

With regard to fission-product sources, a major research effort is still required if the remaining problems of production are to be solved and, if costs of such a program are charged to sources produced, it is improbable that prices can become competitive with those of pile-produced Co^{60} for a considerable time.

TABLE 3—Comparison of Costs Over a Ten-Year Period

Isotope	*Half-life*	*Replacements per 10 yr* (A)	*Source diameter (cm)*	*Flux position × 10^{13} n/cm²/sec*	*Irradiation time (days)*	*Initial irradiation cost* (B)	*Re-irradiation cost* (C)	*Exchange cost* (D)	*Material cost per source* (E)	*Total, (B + E) + (C + D)*	*Average yearly cost (arbitrary units)**
Co^{60}	5.26 y	2	2	3.4	414	14.6	7.3	0.5	0.0	22	2.2
			3	3.4	177	15.6	7.8	0.5	0.0	24	2.4
			4	2.0	183	11.4	5.7	0.5	0.0	18	1.8
			5	2.0	109	10.2	5.1	0.5	0.0	16	1.6
Cs^{134}	2.3 y	4	3	4.3	375	25.4	12.7	0.5	0.0	65	6.5
			4	3.4	208	23.2	11.6	0.5	0.0	60	6.0
			5	3.4	165	20.2	10.1	0.5	0.0	52	5.2
Eu^{152} / Eu^{154}	16 y	0	2	2.0	400	260	—	—	5.0	265	26.5
			3	0.7	618	235	—	—	11.2	246	24.6
Ir^{192}	74 d	52	2	4.3	92	8.64	4.32	0.5	0.4	254	25.4
			3	3.0	65	5.0	2.5	0.5	0.8	159	15.9
			4	2.0	50	4.0	2.0	0.5	1.6	133	13.3
			5	2.0	34	3.55	1.78	0.5	2.3	122	12.2
Sc^{46}	85 d	43	2	3.4	65	2.31	1.15	0.5	1.3	73	7.3
			3	3.4	27	1.95	0.97	0.5	2.9	66	6.6
			4	2.0	25	1.86	0.93	0.5	5.1	67	6.7
			5	2.0	17	1.60	0.80	0.5	8.0	64	6.4
Ta^{182}	115 d	32	3	3.4	194	3.39	1.69	0.5	0.1	71	7.1
			4	3.4	83	2.91	1.45	0.5	1.8	65	6.5
			5	3.4	49	2.61	1.30	0.5	2.8	61	6.1

* When multiplied by 1,000, the cost figures may be taken as very approximate costs in dollars.

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Low-Power Thermal Test Reactor

Designed with a unique internal thermal column, this research reactor allows measurement of relative thermal absorption cross sections to better than 1% by the danger coefficient method. Design criteria and operating data are given

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THE TWO BASIC REQUIREMENTS for neutron physics experiments are a source of neutrons and a detector. Much attention has been given to the design of reactors that provide a neutron flux useful for research or production purposes. Little attention, however, has been accorded the optimum reactor design for detection. One of the primary objectives in the design of the KAPL Thermal Test Reactor was the exploitation of the detection utility of the reactor. That this objective has been met is indicated by the comparison of criteria with characteristics tabulated on this page.

Application of a reactor as a detector involves the measurement of pile reactivity changes resulting from the introduction of sample materials into the active lattice region of a reactor. The measurement may take the form of danger-coefficient experiments† or pile-oscillator experiments.‡ In a danger-coefficient experiment the magnitude of the reactivity change is measured by the displacement of a calibrated control rod that compensates for the poison effect of the sample. In the pile-oscillator experiment the amplitude of the reactor neutron flux variation is measured while the sample is oscillated in and out of the reactor.

To compete with detectors using counting techniques, the accuracy of the reactivity experiments must be comparable to the errors inherent in particle counting. These errors, ranging from 10 to 0.1%, arise from poor counting statistics, counting geometry, or limitations of counting equipment.

In contrast to experiments that use counting techniques for detection, the accuracy of reactivity experiments can be improved by choosing larger samples. And the sensitivity of the reactor as a detector is independent of the absolute neutron flux falling on the sample; consequently the reactor can be operated at relatively low power levels.

Since TTR is designed so that it can be operated at a higher power level, it also serves as a good source of neutrons. In addition to its versatility, TTR has the advantages of being relatively inexpensive and very safe to operate.

Structure of TTR

The physical aspects of TTR are shown in Figs. 1 and 2. The reactor structure proper, shown in Fig. 1, is located in a shielded cell, and can be divided into four regions:

1. Internal thermal column. This is a unique feature of TTR. In the KAPL model a cylinder of graphite 12 in. in diameter and 18 in. high is located in the central hole of the reactor container. A slot approximately 2 in. by $\frac{1}{2}$ in. in the center of this cylinder accommodates experimental samples.

2. Active lattice region. In an annular aluminum container having a 12-in. inside diameter and an 18-in. outside diameter and surrounding the internal thermal column, 20 fuel-slug tubes, each 2 in. in diameter and 24 in. long, are arranged in a cylindrical array. Enriched uranium (greater than 90% U^{235}) is contained in U-Al alloy disks that are about 0.040 in. thick, 2 in. in diameter and have a $\frac{3}{16}$-in. hole through the center.

A fuel slug consists of fuel disks separated by 0.200-in. polyethylene spacers located on an aluminum rod about 18 in. long. The slugs are immersed in a light paraffin base oil in the slug tubes, and the interstices between the slug tubes and the walls of the aluminum container are filled with water.

3. Graphite reflector. The active

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for Nuclear Physics Research

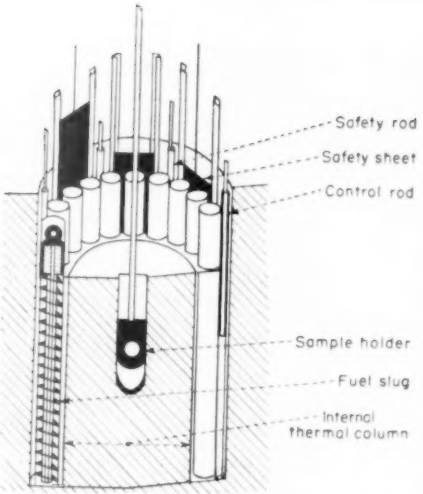

FIG. 1. Core of Thermal Test Reactor

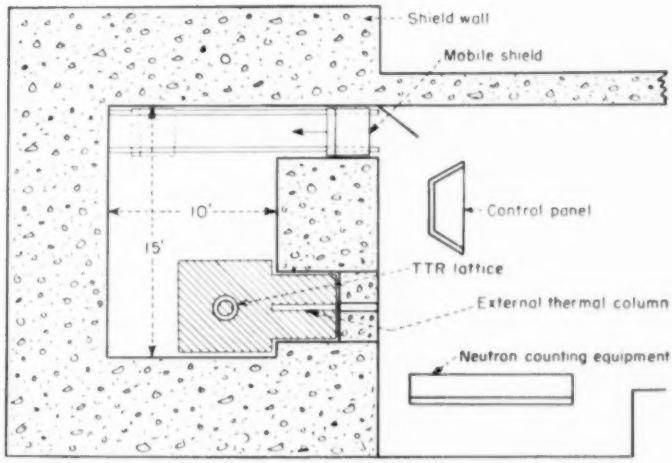

FIG. 2. Floor plan of TTR showing reactor in corner of shielded cell

lattice is surrounded by a graphite reflector that is approximately a cube 5 ft on each side.

4. External thermal column. A graphite cube 4 ft on each side extends partially through the shield wall, as shown in Fig. 2.

TTR is located in a cell that has a floor area of 15 × 10 ft. The cell has a 6-ft poured concrete wall on all four sides, except for a door opening 3 × 7 ft that is closed by a movable shield, and a hole for the thermal column. The ceiling consists of a 1-ft-thick slab of concrete. Attenuation of neutron flux and gamma radiation through the shielding wall is more than adequate for operation of TTR at a power level of 100 watts.

Control room. The reactor is operated by remote controls located on the control panel in the control room. Relay racks are mounted beside the external thermal column to accommodate equipment used in neutron beam experiments.

A Reactor Designed for Detection

These criteria.	. . . are met in TTR by
1. Control system must be sensitive and must provide reproducibility	1. Fractional reactivity change of 5×10^{-7} detected by danger-coefficient procedure.
2. Neutron flux per watt must be large at sample position.	2. Thermal neutron flux at sample position is 3.4×10^{7} n/cm²/sec/watt.
3. Flux spectrum should allow interpretation of experiments.	3. Large ratio of thermal to epithermal flux achieved; cadmium ratio is 5.9 at sample.
4. Critical mass of reactor must be small.	4. About 2.7 kg U^{235} needed to achieve criticality.
5. Sample located so that neutrons absorbed by it are very effective in maintaining pile reactivity.	5. TTR design provides internal thermal column in which there is appreciable augmentation of neutron flux.

Control, Safety Mechanisms

Three sets of neutron-absorbing control rods and plates are used in TTR: coarse controls, fine controls, and automatic safety rods.

Coarse control. Six channels at the periphery of the aluminum container accommodate iron-clad cadmium sheets 4 in. wide and about 18 in. long. These sheets are withdrawn from the reactor individually by remote controls located on the control panel. The total reactivity change effected by the six sheets is about 6.5%. Minimum withdrawal time for all six sheets is about 3 minutes.

Fine control. Two cadmium rods, each 18 in. long and about ½ in. in diameter, are used. These rods move into aluminum guide tubes located between the fuel-slug tubes in the reactor container. The reactivity value for each of the fine controls is about 0.5%. The control rods are driven by motors mounted on a plate above the reactor core. Selsyn systems coupled to the control drives indicate

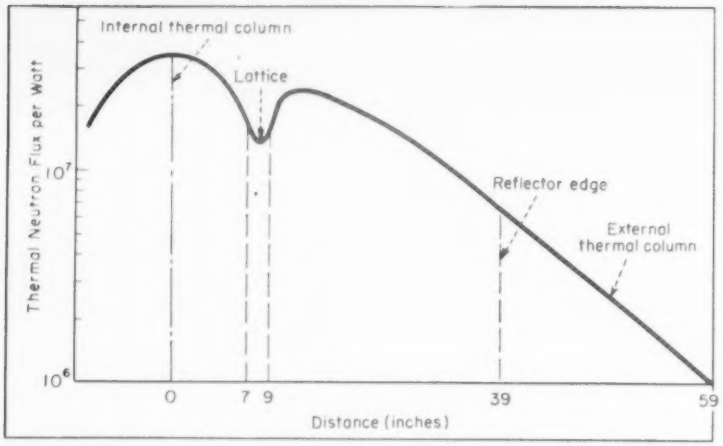

FIG. 3. Thermal-neutron flux intensity per watt in TTR

the rod positions on the control panel.

Automatic safety rods. Four $\frac{1}{2}$-in. diameter cadmium rods, each 18 in. long, are used as automatic safety rods. The total reactivity value of the four rods is about 2.5%.

The rods are suspended above the reactor core by solenoids that automatically release the rods if the power level of the reactor exceeds a predetermined magnitude. The rods and magnetic release mechanism are designed to release the rod in less than 0.005 sec; the rod requires a total of 0.4 sec to drop completely into place after the initial signal. If the reactor should suddenly be brought to prompt criticality, that is, an excess reactivity of about 0.8% (and it is difficult to imagine how this could be accomplished), the power level would increase exponentially with a period of about 0.5 sec. If it is assumed that at least one of the two trip circuits functions properly, the peak power level would not exceed twice the predetermined level for safety-rod release.

Detection Instruments

Neutron and gamma detectors are used for control and health physics during loading and operation of TTR.

Neutron flux and level trip circuits. Two neutron-sensitive paraffin-moderated boron-trifluoride-filled ionization chambers are located outside the graphite reflector; they give signal currents of about 0.03 μa/watt of reactor power. Feedback-stabilized electronic d-c amplifiers operate from these chambers to drive fast-response continuous-writing recorders and high-speed trip circuits. The trip-circuit relays control the power to the safety-rod solenoids and coarse-control-sheet drive motors. The trip circuits are set to operate at about full scale on the indicating recorders. The sensitivity of the electronic amplifiers can be varied by a factor of 10^5 so that the trip circuit is operable down to flux levels of the neutron source alone.

Neutron-counting instrument. Two neutron-counter circuits are used for measuring multiplications during initial loading and for operating an audible indicator. These consist of boron-lined proportional counters driving pulse amplifiers and scalers.

Health instruments. The neutron flux in the assembly room is continuously monitored and recorded by a BF_3-filled proportional counter.

The gamma flux in both the assembly and control rooms is continuously monitored and recorded by an air ionization chamber and electronic amplifier units.

Safety Aspects of TTR

Although experimental versatility and cost of construction were major considerations in the choice of the TTR design, considerable emphasis was placed on safe operational characteristics, including the automatic safety rods.

Interlocks. In the TTR control system, interlocks prevent the beginning of an approach to criticality until the reactor cell is evacuated and the cell door is locked. The coarse control sheets cannot be withdrawn unless power is supplied to the instruments, the fine control rods are in the most poisonous position, the safety rods are cocked, and the neutron source is located in the active lattice region. The fine control rods cannot be moved prior to the complete withdrawal of the control sheets.

Negative temperature coefficient. The light paraffin base oil that surrounds the fuel disks in the slug tubes and serves as a moderator was chosen because of its large bulk modulus and its noncorrosive characteristics.

Since this oil has a large bulk modulus, a small reactor temperature increase causes a large change in the moderator density; this in turn decreases the pile reactivity. If it is assumed that all of the safety mechanisms should fail completely during an accident when the reactor becomes prompt critical (an extremely improbable combination of events), then the strong negative temperature coefficient of TTR would limit the energy release so that the integrated radiation outside the shield would be less than daily tolerance.

Theory of TTR

The performance characteristics of TTR can be described conveniently by applying the danger coefficient theory in a form suggested by Dr. H. Hurwitz, Jr., at KAPL. The reactivity change observed in a reactor due to the addition of a danger coefficient sample is indicated by

$$\frac{\Delta K}{K} = \nu \frac{\Delta f}{\Sigma f} F_H - \frac{\Delta a}{\Sigma f} F_L$$

where Δf = rate of fissions in the danger coefficient sample, Δa = rate of neutron absorption in the sample, Σf = total fission rate in the entire reactor, F_H = favorableness of high-energy neutrons (fission energy spectrum) toward maintaining the chain reaction, and F_L = favorableness of low-energy neutrons (absorption energy spectrum).

If absolute units of reactivity are used, then the favorableness $F(r,E)$ has the significance of an "iterated fission probability" for a neutron of energy E at position r in the reactor. The iterated fission probability is a geometric average of fission probabilities for succeeding generations over an infinite number of generations. The average value of F_H for the entire reactor must be $1/\nu$ (0.4 for U^{235}) for the reactor to be critical. Since fast neutrons have a greater probability for escape from a reactor, the iterated fission probability for low-energy neutrons is usually greater than that for fission-energy neutrons in the active

lattice region of a small reactor. Hence F_L is usually larger than F_H in a reactor core. F_H and F_L are sensitive functions of the position in the reactor.

If the sample consists of U^{235}, then

$$\frac{\Delta K}{K} = \frac{\Delta f}{\Sigma f}[\nu F_H - (1 + \alpha)F_L]$$

where $\alpha = \Delta c/\Delta f$ = parasitic capture rate per fission rate.

If the sample is a nonfissionable absorber, then

$$\frac{\Delta K}{K} = -\frac{\Delta a}{\Sigma f} F_L$$

The more complete expression contains a term describing the effect of scattering on the reactivity. If the sample is located at the center of the reactor, then scattering has no effect on the reactivity. If we write

$$\frac{\Delta a}{\Sigma f} = \frac{nv_0\sigma_a(w_a/A_a)}{nv_{\text{avg}}\sigma_f(w_f/235)}$$

where nv_0 = thermal flux at the center of the internal thermal column, nv_{avg} = average thermal flux in the active lattice region, σ_a = atomic cross section of absorber, w_a/A_a = moles of absorber, σ_f = effective fission cross section of the fuel disks, and $w_f/235$ = total moles of U^{235} in the reactor; then the reactivity change due to an absorber placed in the sample hole is

$$\frac{\Delta K}{K} = -R\frac{\sigma_a w_a 235}{\sigma_f w_f A_a} F_L$$

where R = ratio of thermal neutron flux in the sample hole to that in the active lattice region.

It is noted that the reactivity change is proportional to R and F_L, and inversely proportional to w_f. Hence for a given sample absorber, the maximum reactor sensitivity is obtained by designing the reactor such that the flux augmentation factor R and favorableness factor F_L are as large as possible and the critical mass is as small as possible. The values of F_H, F_L, and R can be found from results of activation and danger coefficient experiments.

Reactor Parameters

Evaluations have been made of the critical mass, power distribution, neutron flux distribution, neutron favorableness, and sensitivity of TTR.

Critical mass. The mass of U^{235} required to achieve criticality in the KAPL TTR was approximately 2.7 kg. An extra 0.1 kg was added for experimental flexibility; it provides about 0.6% in K excess reactivity.

Power distribution. The fission-product activity induced in the TTR fuel disks was measured to determine the power distribution in the active lattice region. The ratio of maximum to average fission rate in the slug is 1.12. A fuel disk which was placed in the sample hole of the internal thermal column showed an induced specific fission rate which was 2.3 times the average specific fission rate in the active lattice region. Hence the augmentation of the neutron flux as measured by U^{235} is $R = 2.3$.

Neutron flux. The thermal-neutron flux at the TTR sample hole was measured by activation experiments with manganese foils. The thermal flux was found to be

$$nv_{\text{th}} = 3.4 \times 10^7 \text{ n/cm}^2\text{/sec/watt}$$

The cadmium ratio of the activation induced in a 0.005-in. indium foil was found to be 5.9 at the sample hole.

Figure 3 shows the thermal-neutron flux intensities per watt along a radius from the center of the internal thermal column through the active lattice, the reflector, and the central axis of the external thermal column. The cadmium ratio for indium foils varies from 100 to greater than 1,000 in the external thermal column.

The epithermal-neutron flux intensity at the sample hole in the internal thermal column has been measured by the self-absorption of strong resonances in indium, gold, and manganese detectors. The foils were calibrated in a graphite sigma pile using a 3-gm radium-beryllium source. The results of the TTR measurements indicate that at 1.44 ev, $Nv = 1.53 \times 10^6$ n/cm²/sec/watt/Δ *ln E*; at 4.9 ev, $Nv = 1.66 \times 10^6$; at 300 ev, $Nv = 1.77 \times 10^6$. The probable absolute error in these measurements is estimated to be 10%; the probable relative error 3%.

Neutron favorableness. The favorableness of thermal neutrons at the sample hole was measured with indium, gold, and manganese foils. The reactivity change, $\Delta K/K$, was measured by the control-rod shift necessary to compensate for the sample, while the absorption per watt, $\Delta a/\Sigma f$, was determined from the activations of the sample foil and monitor disks. Results with the three foils indicated

$$F_L = 0.45$$

The probable absolute error, largely due to inaccurate absolute reactivity calibrations, is 10%, while the probable relative error between the different foil determinations is 2%.

The average value of F_L in the active lattice region is approximately

$$(F_L)_{\text{avg}} = 0.6$$

Although there is an appreciable augmentation of the neutron flux, there is some loss in F_L at the sample position. However, the net effect is an augmentation of the reactivity change by a sample at the sample hole.

Danger coefficient and reactivity measurements of U^{235} at the sample hole position indicated that

$$F_H = 0.42$$

Since F_H is approximately the same as the average value in the active lattice while F_L is somewhat smaller, the sample hole in TTR is particularly sensitive to fissionable samples.

Sensitivity. The value of the reactor as a detector depends not only upon the magnitude of the reactivity change caused by the sample, but also upon the reliability of the reactor control system and the associated instruments for measuring the reactor power level. Experience with TTR indicates that a reactivity change of 5×10^{-7} in K can be detected by normal danger-coefficient procedures. With some improvement in the instrumentation it should not be difficult to detect a reactivity change of 10^{-7}. A 1.5-gram gold foil having a macroscopic absorption cross section of approximately 0.4 cm² causes a reactivity change of $\Delta K/K = 2 \times 10^{-4}$ in TTR. Hence a cross section of 10^{-3} cm² can be measured by routine danger-coefficient methods. Improvements in the instrumentation might decrease the minimum detectable cross section by a factor of 5. A pile oscillator might give greater sensitivity.

The limit to detection sensitivity appears to be instability of the reactor due to temperature changes. A temperature change of 10^{-3}° C causes a reactivity change of $\Delta K/K = 10^{-7}$. Consequently, it is important to maintain good temperature control.

A convenient size for danger-coefficient samples in TTR is one which causes a reactivity change of $\Delta K/K = 10^{-4}$ to 10^{-3}. Hence it should be practical to measure relative thermal absorption cross sections with an accuracy better than 1%.

* * *

The authors wish to acknowledge the contributions of E. E. Haslund and P. L. Clairmont in design and assembly of the reactor.

Neutron Cross Sections in the Mev Region

To continue the expansion of reactor technology, much work on neutron cross sections is needed. Progress and problems in this work, with emphasis on the needs of AEC, were discussed at the conference reported here

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A CONFERENCE on neutron cross sections in the million-electron-volt region was held at Brookhaven National Laboratory, Upton, New York on January 20 and 21, 1953. Sponsored jointly by the ONR and the Neutron Cross Section Advisory Group of the Atomic Energy Commission, its purpose was to bring together those people actively engaged in the measurement of neutron cross sections to discuss progress and problems in this field. Special emphasis was placed on the cross section needs of the AEC.

The outline of this report of the proceedings follows the program of the conference. Only four invited talks were given (by Weisskopf, Goldstein, Feshbach, and Muehlhause); the rest of the sessions were devoted to informal discussion.

Total Cross Sections

V. F. Weisskopf (MIT) opened the first session with a review of the present status of the theory of total cross sections. In the light of recent experimental data, some of the theoretical concepts developed during the past 10 or 15 years about nuclear reactions do not seem to stand up. The success of the shell model in explaining other nuclear phenomena suggests its application to the theory of total neutron cross sections.

The recent work of Barschall and his co-workers at Wisconsin (*1*) has shown experimentally that total cross sections for medium and heavy elements do not decrease monotonically with energy but show a broad peak which shifts to higher energies with increasing atomic weight. The standard theory of nuclear reactions (*2*) predicts a monotonic decrease. This theory is based on the strong-interaction, Bohr picture of compound-nucleus formation, in which it is assumed that the instant the neutron enters the nucleus it becomes amalgamated with the other nucleons because of the strong interaction between nucleons.

The success of the shell model (or independent-particle model) at low excitation energies casts some doubt on the strong-interaction concept and suggests a relaxation of this assumption. If the neutron is allowed a mean free path in nuclear matter of two or three times the nuclear radius (for $A \sim 100$), excellent agreement can be obtained with the experimental curves of the Wisconsin group. This model indicates only the "coarse structure" in nuclear reactions. The "fine structure" (resonance structure) would appear if the compound nucleus and its decay were considered in more detail.

Low-energy resonances. It has only been within the past year that it has been possible to make neutron cross section measurements in the energy region from 1 to 10 kev with sufficiently good resolution to reveal resonance structure. This has been accomplished by the development of techniques that allow the extension of the useful range of Van de Graaff generators down into this region and, simultaneously, by the refinement in high-speed rotating-shutter velocity selectors which utilize the high-energy neutrons available in a reactor spectrum.

A. S. Langsdorf, Jr., (Argonne National Laboratory) described the work done in extending the range of Van de Graaffs to these low energies (*3*). Previously, most work stopped at about 15 kev, with quite poor resolution at this energy. Lower-energy neutrons are available at large back angles from the Li(p,n) reaction, but low neutron yield and high background counting rates have made it difficult to tap this source. By using a massive shield to reduce background and a much more efficient ($\sim$10%) detector, it has been possible to work down to $\sim$2 kev. It should be possible to go down to neutron energies of 500 ev, with a 500-ev resolution, using a lithium target with $\sim$1-kev stopping power.

Similar work was reported by H. W. Newson (Duke University). Preliminary studies have been made of the 2.5-kev resonance in sodium.

Extension of mechanical velocity selectors (fast choppers) into this region was discussed by D. J. Hughes (Brookhaven National Laboratory). At 1 kev, the energy resolution of the chopper (about 10%) is better than that available with the Van de Graaff generators; at 5 kev, they are about equal. A resonance in manganese at about 1 kev has been found at Brookhaven.

Cross section measurements. Recent total cross section measurements, utilizing Van de Graaff generators, were discussed by R. L. Henkel (Los Alamos Scientific Laboratory), C. H.

Johnson (Oak Ridge National Laboratory), and S. C. Snowdon (Bartol Research Foundation). The Los Alamos work was in the region from 1 to 8 Mev and included investigations of helium, carbon, oxygen, aluminum, and uranium. Much new resonance structure was found. Some work was also reported in the region from 14 to 21 Mev, and an additional maximum in the cross section of uranium was found in this region.

At Oak Ridge, an extensive study was made of Li^6 (using a lithium sample enriched to 86% Li^6) from 35 kev to 4.2 Mev. The level at 258 kev was found to have a width of approximately 100 kev and a peak height of 10 barns.

In another experiment, angular distributions of neutrons scattered by nitrogen in an ion chamber were measured, using the pulse-height distribution of the recoiling nitrogen nuclei. At some resonances, in the region from 1.3 to 2.0 Mev, much forward scattering was found; at others, much back scattering.

The Bartol work was done in the region from 100 to 700 kev, with a resolution of ~40 kev. Some resonance structure was revealed in chlorine and phosphorus; gold was also investigated.

The measurements by R. W. Lamphere (Oak Ridge National Laboratory) of the fission cross sections of U^{234} and U^{236} were reported by W. M. Good. The measurements were made relative to the fission cross section of U^{235}. Thick samples were calibrated by means of thin samples and a Po-Be neutron source. The chief problems lay in the foil preparation and knowledge of foil thickness.

Neutrons from the Los Alamos fast reactor have been used to make total neutron cross section measurements in the region from 3 to 13 Mev. C. W. Zabel discussed the work of N. Nereson and S. Darden (*4*), who used a hydrogen-recoil ionization chamber, with an energy resolution of approximately 10%, as a detector. About a dozen elements were investigated, the heavier elements showing the same regularities in the average cross sections observed earlier by Barschall. The correction due to the scattering of neutrons by the sample into the detector was very small in this experiment, and a comparison with overlapping Wisconsin results indicated discrepancies up to 10%. If an inscattering correction due to strong forward scattering were applied to the Wisconsin data, these discrepancies would probably disappear.

Additional work with this experimental setup is being carried out at the Los Alamos water boiler with a thermal-neutron converter. Such a converter consists of a plate of U^{235} irradiated with a well-thermalized beam of neutrons, the thermal neutrons causing fissions. The thermal-neutron flux is thus "converted" into a fission flux. This new arrangement provides an increase by a factor of two in the available neutron flux.

Inelastic Scattering

The session opened with a talk by H. Goldstein (Nuclear Development Associates) on the measurement of unclassified fast-neutron cross sections of interest to AEC for shielding and reactor studies. A complete article on this talk will be published in an early issue of NUCLEONICS.

Spherical-shell method. One of the more practical and productive techniques for measuring total inelastic collision cross sections (total cross section minus elastic scattering cross section) has been the method of spherical shells. In this method, a thin shell of the material under test is placed around either the source or the detector, the one not surrounded being some distance away.

The advantage of this method is that, for sufficiently thin shells, the effect of the elastically scattered neutrons essentially cancels out. This is because the number of neutrons scattered into the detector per second is equal to the number of neutrons per second scattered out of the direct beam from source to detector, assuming the scattering is isotropic.

Detection is chiefly done with threshold detectors. What is then measured is the cross section for scattering to an energy below the threshold energy of the detector. Photographic plates and other energy-sensitive detection methods can also be used.

Considerable work has been done at Los Alamos with the sphere technique. R. C. Allen described measurements of total collision cross sections using neutrons of energies from 0.2 to 1.7 Mev and U^{238} and Np^{237} fission-threshold detectors. Shells of 8- to 10-cm diameter were used to surround the detector, and the source was 10 cm away.

A high-pressure recoil proportional counter is also being developed. The cross sectional dimensions of the chamber are $1\frac{1}{4}$ by $\frac{15}{16}$ in., with an active volume of about $\frac{5}{8}$ in. along the wire and $\frac{1}{4}$ in. around it. The pulse-height distributions must be further refined, but preliminary results are in agreement with the threshold-detector results.

Measurements made with a fission-spectrum neutron source were presented by R. E. Carter. The thresholds used included the U^{238} and Np^{237} fission thresholds and the $Al(n,p)$ threshold. What was actually measured was the total inelastic collision cross section, averaged over a fission spectrum, times the fission cross section of the threshold detector.

G. M. Frye reported on some measurements made by Graves and Rosen (*5*) with an intense 14-Mev source and photographic plates as detectors. The spheres in this case were around the source rather than around the detector. The neutron distribution showed the large peak at about 14 Mev, corresponding to neutrons transmitted and scattered elastically, and a tail at very low energies which seemed to increase in height as Z increased and continued to rise down to the lowest energies observed.

Inelastic scattering in iron. An experimental measurement done at MIT on inelastic scattering in iron (*6*) was described by P. H. Stelson (now at Oak Ridge). A borated-paraffin collimator was used to shield the photographic plate detectors from the direct beam. The track density in the photographic plates was quite low, and the statistics were therefore very poor.

A similar experiment is planned at Oak Ridge as a result of a suggestion by Barschall. If the well-known $T(p,n)$ reaction is reversed, and 4.5 Mev tritons are allowed to bombard hydrogen, a beam of ~2-Mev neutrons would be obtained in a forward cone of about 35-deg half-angle. This would give a gain of a factor of 8 in intensity, thereby markedly increasing the track density and statistics.

Inelastic scattering theory. H. Feshbach (MIT) began his talk on inelastic scattering by assuming the Bohr strong-interaction model of the nucleus. He then discussed the results of the relaxation of this assumption as a consequence of the new ideas introduced

by Weisskopf. If the statistical theory can be applied to the compound nucleus, it is possible to derive an expression for the inelastic scattering cross section that is dependent upon the energy and angular momenta of the incoming and emitted neutrons.

For neutrons near threshold energy for a particular level in the residual nucleus, the shape of the cross-section curve will be very sensitive to the angular momentum of the incoming neutron. Thus, this shape can give some idea of the spin change. The angular distribution above threshold is anisotropic, but it never seems to become strongly so. If one can apply statistical theory to the residual nucleus as well as to the compound nucleus, the angular distribution will be isotropic.

The energy distribution of the inelastically scattered neutrons will be approximately Maxwellian. A high-energy tail will be superimposed upon the Maxwellian if the residual nucleus excitation energy gets too low (~3 Mev).

If the neutron leaves the residual nucleus in a high enough state of excitation, the emission of another low-energy neutron is possible, i.e., an $(n,2n)$ reaction may occur. This would distort the Maxwellian spectrum of the first emitted neutrons strongly towards low energy, the exact shape depending upon the low-lying levels. If a constant low-energy level density is assumed, the maximum will appear at zero energy. This might be an explanation of the low-energy tail reported earlier by Frye.

If Weisskopf's new interaction concepts are introduced, there will not be a radical change in the total inelastic cross sections, which are quite independent of model. The major change will come in the angular and energy distributions. If an experiment were performed such that only one level in the residual nucleus were excited, the shell model and the new picture would predict a sharp anisotropy, an absence of symmetry about 90 deg, and an increase in the number of neutrons in the high-energy tail of the spectrum.

Detector shielding. The problem of shielding detectors from the direct beam in an inelastic scattering experiment was discussed by C. E. Mandeville (Bartol). A study of the best geometry to shield photographic plates was made with two different sized paraffin wedges, a bismuth wedge, and a Stelson-Preston type paraffin collimator as the possible shielding materials.

It was found that the small and large paraffin-wedge geometries were as effective as the Stelson-Preston collimator, so a paraffin-wedge geometry was adopted which allowed the retaining of all pulses greater than a certain value. The same geometry should be applicable if scintillation detectors are used in place of the photographic plates.

Gamma-ray observation. One of the best methods of studying inelastic scattering is by direct observation of the gamma rays emitted in the process. Such an observation was reported by R. B. Day (Los Alamos) in which a scattering ring of the material being studied surrounded a NaI(Tl) scintillation detector which was shielded from the direct neutron beam by means of a tungsten cone.

The gamma rays were measured by pulse-height analysis of the detected radiation. Known transitions in Be, B^{10}, C, F, Mg^{24}, Al, S^{32}, and Fe were observed. The absolute gamma-ray yield was determined with the four well-known gamma rays from Na^{22} and Na^{24}. By calibrating the neutron flux, inelastic cross sections could be calculated perhaps to about 10 or 20%.

Detectors and Instrumentation

C. O. Muehlhause (Brookhaven) began his review of short-memory radiation detectors by noting that such detectors may be classified according to whether the medium involved is a gas, liquid, or solid and according to whether charge, light, or sound quanta are the fundamental forms of energy which are integrated into a pulse. Sound was introduced since a counter has been suggested in which the piezo effect and fission are utilized.

The different detectors can also be classified in terms of their detecting properties. Some such properties are efficiency, energy resolution, background sensitivity, and response time. Only liquid and solid media, containing hydrogen, have the possibility of good efficiency.

Good energy resolution is possible if sufficient pulse magnitude is obtained to permit differential analysis or if the detector time is sufficiently fast to permit time-of-flight measurements. Background sensitivity will be low if a low-density, low-atomic-number medium is used, and this only if either pulse-height-analysis or time-of-flight measurement is possible.

Response time is either of the order of microseconds, if electron collection or slow phosphors are involved, or of the order of millimicroseconds, if fast phosphors are involved. For a detector to be suitable for time-of-flight measurements of Mev neutrons, its response time must be of the order of millimicroseconds. A good fast-neutron detector must possess good efficiency, low background sensitivity, and either good energy resolution or fast time response.

The best all-purpose counter seems to be of the fast, boron-loaded, liquid-organic-scintillator type. Such counters double-pulse owing to proton recoil plus delayed boron capture, and thereby permit discrimination against gamma rays and slow neutrons. Energy resolution via pulse height is modest.

Pulsed sources. The problem of obtaining pulsed sources of fast neutrons was discussed by several experimenters. Time-of-flight techniques should be very effective in the measurement of inelastic cross sections, but this method is only possible where a source of pulsed neutrons is available.

At Westinghouse, as described by B. Jennings, a method has been devised for using oscillator-fed deflecting electrodes to pass the deuteron beam from an electrostatic generator into the entrance slits of an electrostatic analyser. The analysed pulse was led on to a deuterium gas target and neutrons were produced. The pulses from the terphenyl-xylene or stilbene detector were displayed on an oscilloscope. The over-all time resolution for 4-Mev neutrons and a 1-m flight path from target to detector should be better than 1×10^{-9} sec.

A similar approach is being developed at Oak Ridge. C. W. Snyder reported that a larger-area scintillation detector and larger distances will be used so that longer neutron pulses can be tolerated. The sweep on the oscilloscope will be triggered by the inelastically scattered neutrons so that times will be measured "backwards," i.e., time from inelastic pulse to next timing pulse.

Another scheme for producing short intense neutron pulses was proposed by R. C. Mobley (7) (Duke). A pair of deflectors at right angles to each

other are fed by two phased oscillators, one with a frequency three times the other. One pulse from the electrostatic generator is therefore passed into the focusing magnet for each three cycles of the higher-frequency oscillator. The deflecting magnet bunches a 2.6-mm-wide beam of 5-Mev protons into pulses of the order of 2.6×10^{-10} sec, thereby intensifying the peak current by a factor of 50. For a 1-m flight path and a 4.3-Mc repetition rate it is possible to work down to a neutron energy of 100 kev. The neutron spectrum could be obtained from the fast coincidence of the counter pulse and the pulse for the target, with a variable time delay in the target pulse.

Preliminary tests have been made on obtaining pulsed neutrons from the Cornell 16-in. cyclotron, and a scheme has been suggested for achieving pulses from the Los Alamos cyclotron. J. E. Draper (now at Brookhaven) described the work done at Cornell. The natural time bunching of the 100-kev deuteron beam was used to produce pulses of 14-Mev T(d,n) neutrons. Fast amplification of the beam current pulse was used as a timing pulse for arrival of the beam. A stilbene crystal was used as a detector for the neutrons, the full pulse width being 5×10^{-9} sec with a repetition rate of 8 Mc. The flight path was 60 cm, and time-of-flight measurements were made. Some inelastic pulses of several Mev energy were found in the time-of-flight distribution when the target was surrounded by 10 cm of aluminum.

K. Boyer (Los Alamos) discussed the possibility of utilizing the phase bunching in the cyclotron to produce pulses. Pulses about 8×10^{-9} sec wide should be obtained since the cyclotron operates at 11.5 Mc, and the accelerated particles are bunched to about a 30-deg spread, and this has been checked experimentally. Reducing the spread to 5 or 6 deg should yield a factor of 5 in pulse width, but the average beam current would probably decrease. Neutrons from 1 to 100 kev could be obtained by observing singly-scattered neutrons at right angles to the deuteron beam, and any spreading of the burst would be eliminated by this procedure. Experimentally this is done by placing a thin paraffin slab behind a beryllium target.

The use of strong focusing, originally suggested for a high energy synchrotron (*8*), can also be applied to low-energy accelerators. J. H. Williams described application of strong focusing to an accelerator at Minnesota. Two hyperbolic lenses, 3 in. long with a 2-in. gap, separated by 6 in. and operated at 7 kv are adequate to focus a 0.5-Mev proton beam. Berkeley has succeeded in focusing the proton beam in a linear accelerator, and the strong-focusing principle seems to work beautifully.

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Other Methods

In addition, spectrometers, scintillating detectors, cloud chambers, and the Ranger method were discussed.

Spectrometer. A He^3-counter to be used as a neutron spectrometer for inelastic scattering measurements in the region from 0.4 to 1.5 Mev neutron energy was described by B. J. Toppel (Brookhaven). The sensitive volume was 2 in. in diameter and 5 in. long. Filled with <0.5 cm³ of He^3 and 6 atmospheres of argon, the counter has been tested with thermal neutrons. An energy resolution of 4.5% was obtained for the 0.76-Mev pulses produced by the $He^3(n,p)$ reaction.

When the counter was tested with fast neutrons from the D(d,n) reaction, an apparent nonlinear variation of pulse height with energy was found. This effect was explained by the presence of an (n,α) reaction in argon which has a Q value of 1.8 Mev. Krypton will therefore be used in place of the argon.

Scintillating detectors. An extensive study of lithium-containing solid phosphors has been carried out at Oak Ridge in an attempt to obtain better scintillating detectors for fast-neutron detection and energy measurements. Several oxygen-containing phosphors, including silicates, titanates, phosphates, zirconates, stanates, etc., with various activators, were investigated with ultraviolet light and polonium alpha particles.

Only Li_2CaSiO_3, Li_2SnO_3 activated with titanium, and the zirconates had decay times which made them seem promising. LiI phosphors, containing such activators as SnI_2, TlI, and Eu, were also tested with thermal neutrons, fast neutrons, and gamma rays. The best results have been obtained with lithium iodide activated with europium. The increased pulse height obtained means improved resolution. The light output was approximately a linear function of particle energy, as tested with various energies of gamma rays. Preliminary tests on LiF with 1% LiI activated with europium looked encouraging.

Cloud chamber. The usefulness of a diffusion-type cloud chamber for survey purposes in inelastic scattering has been demonstrated by W. B. Fowler (Brookhaven). A collimated beam of 3-Mev neutrons passed through an 18-in. cloud chamber filled with 10 atmospheres of hydrogen, in the center of which a carbon scatterer could be placed. In spite of a background of source neutrons, the tracks from the carbon scatterer were several times background. By setting rigid criteria on the tracks with regard to energy and angle, a resolution of 30% was obtained. Although the actual operating time at a machine was small, the scanning was tedious and very slow.

Ranger method. The Ranger method of Hill (*9*) consists of a thin paraffin layer, which serves as a source of recoil protons, and a detector, which consists of three coincidence counters and one anticoincidence counter. The range of protons stopped in the detector is selected by adjusting the absorptive path between the paraffin source and the detector.

D. A. Cowan (Vanderbilt University) suggested this method as a useful neutron spectrometer with low background but very low efficiency. It has already been used to measure the energy distribution of fission neutrons (*9*) and the neutron spectrum from a Ra-Be source.

* * *

The contribution of Dr. John A. Harvey in the preparation of this review was considerable and is gratefully acknowledged.

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Many shortcomings of radon seeds for medical use are overcome with the Au^{198} seeds described here. A simple fabrication method has been developed, and refinements have been made in the Nylon-tube implantation technique

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FIG. 1. Encased radiogold wire is extruded from lead shield. Meter at left indicates strength; slight blow on plastic head cuts seed, which falls through tube into shielded container where electric heater sterilizes seed

Radiogold Seeds for Cancer Therapy

RADIOGOLD SEEDS and linear sources developed during the past three years (*1–5*) have displaced radon completely in radiation therapy at The Ohio State University Medical Center. These sources are now used advantageously in many types of cases in which radioactive cobalt was formerly applied (*6–10*).

In external appearance the Au^{198} seeds are indistinguishable from the usual radon seeds, and they can be utilized in the same manner and with the same apparatus. The half-life of Au^{198} is 2.70 days, which is about 0.7 of the half-life of radon (3.83 days). The energy of the gamma radiation emitted by Au^{198} is 411 kv; it is monoenergetic in contrast with the range of energies between 184 kv and 2,198 kv for the dozen different gamma rays emitted by radon and its disintegration products (*11*). Further biological studies (*1, 2*) are under way to determine the relative merits of the differences in half-lives and gamma-ray energies of Au^{198} and radon.

Sources for interstitial applications should not be confused with radioactive gold in the colloidal state which is now being used in some medical centers for the treatment of pleural and peritoneal carcinomatosis. In contrast to colloidal Au^{198}, in which the beta radiation emitted by the radiogold is used primarily, gold seeds containing Au^{198} are employed only because of the

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gamma rays emitted from the radiogold within them. The seeds are usually superior because of the relatively homogeneous dosage of ionizing radiation that may be achieved when discrete sources of gamma rays are properly spaced in accordance with well-established physical principles. Dosimetry is highly uncertain whenever radioactive sources are injected that are mainly dependent on the emission of beta particles for whatever effectiveness they may have. The Au^{198} seeds are, therefore, essentially similar to radon seeds, where much of the beta radiation is removed by filtration.

Advantages of Au^{198} seeds over radon seeds for applications in therapy include:

1. Radiogold seeds of any strength can be cut instantly, even at the time the decision to use them is made (e.g. in the operating room), whereas radon seeds usually must be ordered a day or more in advance of the time they are to be used.

2. Radiogold seeds are very uniform in strength, but wide variations in strengths are often found in radon seeds.

3. Because gold is a metal there is no danger of leakage of gas from Au^{198} seeds.

4. The beta particles emitted by Au^{198} with maximum energy of 970 kv are almost completely filtered out, in contrast to the beta particles emitted by radon disintegration products with maximum energy of 3,170 kv.

5. The facilities and equipment required for manufacturing seeds loaded with Au^{198} are simple and inexpensive.

6. Problems of protection are greatly simplified in the preparation and handling of Au^{198} seeds because the half-value layer in lead for the gamma rays is only 2.8 mm, in contrast to 14 mm for radon.

Preparation of Sources

Each Monday there arrives in Columbus via air express a package shipped by the Oak Ridge National Laboratory earlier that day. This package contains several pure gold wires† 0.2–0.3 mm in diameter and 85 mm long that have been irradiated during the previous week in a flux of about $1–5 \times 10^{11}$ thermal neutrons/cm^2/sec in a nuclear reactor. In the radioisotope laboratory the radioactive wires are slid into inert gold tubing† about 0.4 mm inside and 0.8 mm outside diameter. This nonradioactive outer gold tubing absorbs almost all of the beta particles and passes about 90% of the 411-kv gamma rays emitted by the radioactive gold wire contained within it.

Seeds are cut to any desired strengths with a simple cutter. The device consists of a lead block with a central bore into which the encased radioactive gold wire is introduced and from which it may be extruded to any desired length from 1 to 40 mm, preadjusted by a micrometer. The gamma radiation of the part of the encased radioactive gold wire pushed outside of the lead shield to be cut off with the cutter, is measured with a radiation survey meter equipped with an ionization chamber ("Cutie-Pie" type).

FIG. 2. Six Nylon tubes were used to implant radiogold seeds in residual tumor at base of tongue after radical surgery. Dummy seeds used first, then replaced with active seeds to give minimum tumor dose of 5,200 r in 7 days

Calibration of seed strength. The survey meter is placed in a fixed position at such a distance from the cutter that a swing of the needle to 20 scale divisions on the 50 mr/hr full-scale range of the meter will result in a Au^{198} seed which will deliver 1,000 gamma roentgens-total-at-one-centimeter (designated "rtcm") during complete decay. This is approximately the same strength as the 1,112 r given off during lifetime decay of 1 mc of radon enclosed in 0.5 mm. of platinum. The cutter is portable and is often taken to the operating room where Au^{198} seeds are calibrated and cut in a few seconds. (See Fig. 1.)

Details of design of the cutter, of the method of calibration of the sources, and of dosimetry with the Au^{198} seeds will be published separately.

Following publication (*1, 2*) of the method developed here for making sources containing Au^{198}, Sinclair (*12*) briefly described sources made from cylinders of gold sealed in a platinum beta-particle absorber before exposure to the neutron flux in a nuclear reactor. The method described here for preparing Au^{198} seeds would appear to be a superior one in the following respects:

1. The radiation therapist has all of the advantages of on-the-spot adjustability of strengths of sources provided by a radon plant without the many well-known disadvantages associated with the establishment and operation of a radon plant. The method described by Sinclair does not permit adjustment of strengths of sources at the point of usage, nor does it enable the therapist to compensate for decay of radioactivity by making longer seeds or by using successively thicker radioactive gold wires irradiated in the same container in the nuclear reactor, during the latter part of the week they are received.

Radiogold seeds received from a distant point of preparation will always put the therapist to the same disadvantages that now prevail for those physicians who receive radon seeds from commercial sources. The variation of source strengths to provide the flexibility required for individualization of patterns to meet the multitudinous clinical variations encountered in practice, is not possible with sources of initially fixed strengths.

2. It was shown previously (*3, 5*) that the leakage of the undesirable beta particles through the thinned parts of the ends of Au^{198} seeds prepared here is insignificant in comparison with the escape of highly energetic and penetrating beta particles from a radon seed obtained from a commercial source. Measurements made by Sinclair revealed a beta-particle emission from the platinum-sheathed Au^{198} sources he prepared that differed only slightly from that of radon seeds.

3. The cost of nonradioactive gold for sheathing material to absorb the

† Supplied by the American Platinum Works, Newark, New Jersey.

same fraction of the beta particles emitted from the enclosed radiogold is only about one-fourth as great as that for the platinum sheaths used by Sinclair. Moreover, the expense incurred for the precision machining and welding that is required to encapsulate the gold within the platinum is avoided in the method devised here.

Clinical Uses

Radiogold seeds may be utilized to form permanent implantations by means of the same applicators that are now widely used for radon seeds. Alternatively, Au^{198} seeds may be loaded into Nylon tubing that is threaded through tumors and withdrawn at the end of the exposure time, which is usually 7 days. An advantage of implanting Au^{198} seeds permanently is that it is unnecessary to load the seeds into Nylon tubing. However, permanent implantations by free-hand techniques have not resulted in the regular patterns of sources that are usually obtained readily when the seeds are first enclosed in Nylon tubing.

Another advantage of the Nylon-tube method is that errors in alignment of the pattern during implantation can be corrected by pulling the Nylon tubes into the proper position, as indicated by roentgenograms. Also, no foreign body remains in the tissue after the Nylon tubes are withdrawn at the end of a week. Experiences at this medical center have demonstrated that the Nylon-tube technique is generally to be preferred to the permanent implantation of the seeds.

Improved implantation method. The Nylon-tube implantation method, previously developed here for use with Co^{60} (*8–10*), has recently been improved in several respects. Since details of this new Nylon-tube technique will be reported independently by Henschke, only the principal features of it will be outlined briefly here. First, in place of the thick-walled, relatively stiff Nylon tube formerly employed at The Ohio State University Medical Center, a thin-walled, narrow Nylon tube is now used. In this new Nylon tube, the radioactive sources are held firmly in place by the resiliency of the Nylon without the need for the metallic spacers used formerly. These new tubes are much more flexible and can be bent more readily than the previous tubes. Moreover, should the loaded suture break inadvertently, the radioactive sources cannot fall out as they could with the old method.

A second major improvement in the Nylon-tube technique has been evolved and is now used in many cases. The Nylon tubes are not loaded with the radioactive sources prior to implantation; instead, Nylon tubes loaded with dummy metal seeds, or thin stainless-steel wires are inserted in the tumor area at the time of operation. Roentgenograms are then made and an appropriate distribution for the radioactive sources is determined from the spatial relationships. Nylon tubes are loaded with the radioactive sources accordingly and are pulled into position after attachment to the dummy-loaded Nylon tubes or to the stainless-steel wires. This substitution of the Nylon tubes loaded with active sources for the ones containing the dummy sources, or the wires, causes little discomfort and is usually done satisfactorily without anesthesia. (See Fig. 2.)

It will be apparent that this technique enables the surgeon to prepare a patient during operation for subsequent implantation with the radioactive seeds. The radiation therapist can then design an appropriate pattern and insert it at a time and place convenient for him. This technique has proved mutually satisfactory to the surgeon and the radiotherapist members of the team here.

Clinical experience. Fifty-one applications of Au^{198} seeds clinically in 46 patients thus far have included 31 permanent implantations, 12 Nylon-tube implantations, 2 intrauterine applications, 2 intraesophageal applications, 2 nasopharyngeal applications, 1 intraoral mold, and 1 skin mold.

No untoward or unusual reactions have been noted in any case in which the Au^{198} seeds have been used. The clinical regressions of tumor tissue have been comparable to those observed with other methods of interstitial irradiation. And they have been consistent with the preliminary animal experiments in which it was demonstrated that it is possible to deliver a cancerocidal dose to tumor tissue with Au^{198} seeds alone (*1, 2*). However, the preferred practice here usually is to use Au^{198} seeds in combination with high-voltage external irradiation. In some cases two insertions of radiogold seeds spaced 1–8 weeks apart have been used to deliver the total dose over a longer period.

Summary

1. Radiogold seeds are convenient replacements for radon seeds. And they may also be used advantageously in the newer methods of application such as the Nylon-tube implantations.

2. Radiogold seeds are prepared by activating a fine gold wire in a nuclear reactor and then encasing it in an inert outer gold tube in order to filter out the beta particles. The Au^{198} seeds are cut with a special cutter to any desired strengths registered on a radiation meter as the encased radioactive gold wire is extruded from a lead storage shield. The meter is calibrated to give the number of gamma roentgens-total-at-one-centimeter, rtcm, which the seed to be cut will emit during complete decay.

3. The reactions observed clinically in 51 applications in the first 46 patients in which the Au^{198} seeds were used appeared to be similar essentially to those obtained with implantations of radon, radium, or Co^{60}.

* * *

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Photovoltaic Radiation Detector

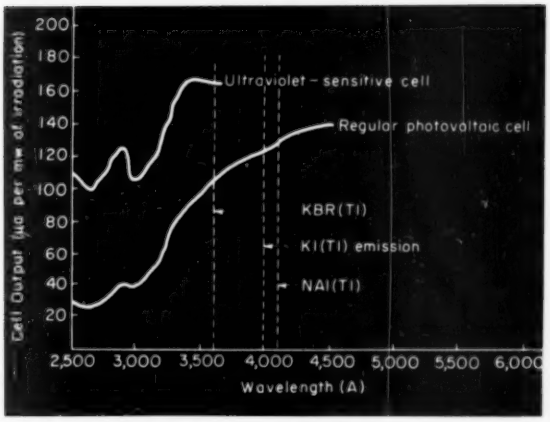

FIG. 1. Sensitivities of "violet-sensitive" and "regular" photovoltaic cells and principal phosphor emission wave lengths

A sandwich detector, comprising two photovoltaic cells and a NaI(Tl) scintillator, can monitor gamma radiation fluxes of 10 r/hr or higher. Sensitivities have been attained of 0.02 μa/r/hr for 140-kv X-rays and 0.0056 μa/r/hr for Ra γ-rays

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THE PHOTOVOLTAIC RADIATION DETECTOR comprises a phosphor placed in close proximity to one or more photovoltaic cells. When the phosphor is stimulated by nuclear radiation, it emits light which falls on the photovoltaic cell; the amount of cell current, indicative of the intensity of incident nuclear radiation, is usually read on a microammeter or galvanometer. The cell current decreases as the resistance of the external load increases.

The detector needs no batteries, vacuum tubes, or other auxiliary equipment. Since none of the components of the detector are delicate, the complete unit can be very rugged. If the sensitivity requirements are not too severe, the detector can be made to small dimensions. The smallest made to date is a cylinder 1 in. in diameter and 1 in. high. Because of the low impedances involved, remote readings may be made very easily; no complications from pick-up occur if the two signal leads from the detector are extended a few or thousands of feet to meet the recording meter.

Although the detector is designed to measure relatively high gamma fluxes, all the components preferably should have as great a sensitivity as possible. The detector may then be made to smaller dimensions.

Phosphor and Cells Used

For the detection of gamma rays, a heavy (high atomic number) phosphor would be generally desirable (*1*). The literature (*2, 3, 4*) suggests alkali halides and cadmium sulfide for use with such radiation. The latter is difficult and expensive to produce. Of all the alkali halides tested for use in this detector, thallium-activated sodium iodide proved the best in terms of current output for a given incident radiation intensity. The phosphor, in the form of clear crystalline pieces, is supplied by the Harshaw Chemical Company, Cleveland, Ohio.

The photovoltaic cells are the product of the Meter and Instrument Division of General Electric Company at West Lynn. These cells are available as "regular" or "violet-sensitive" cells. Their sensitivities are illustrated in Fig. 1, together with the principle emission wave lengths of some alkali halides. That of NaI(Tl) occurs at about 4,100 Å.

Detector Construction

The detector used in the tests described here utilized the photovoltaic cell type PV7 (regular cell), which is approximately 4 in. in diameter and is about $\frac{1}{16}$ in. in thickness. Approximately 175 gm of phosphor were used. Figure 2 shows a prototype unit and its cross section. The case is of textolite plastic. The phosphor is sandwiched between two P.V. cells, with mineral or silicone oil surrounding the components for better optical contact.

Preliminary investigations were made of several phosphor-cell configurations (Fig. 3):

1. A single P.V. cell placed on the side of the phosphor away from the X-ray source.

2. A backing reflector of 0.2-mil aluminum foil placed on top of the phosphor in arrangement 1.

3. The sandwich arrangement.

The lower output of the sandwich arrangement under X-rays is caused

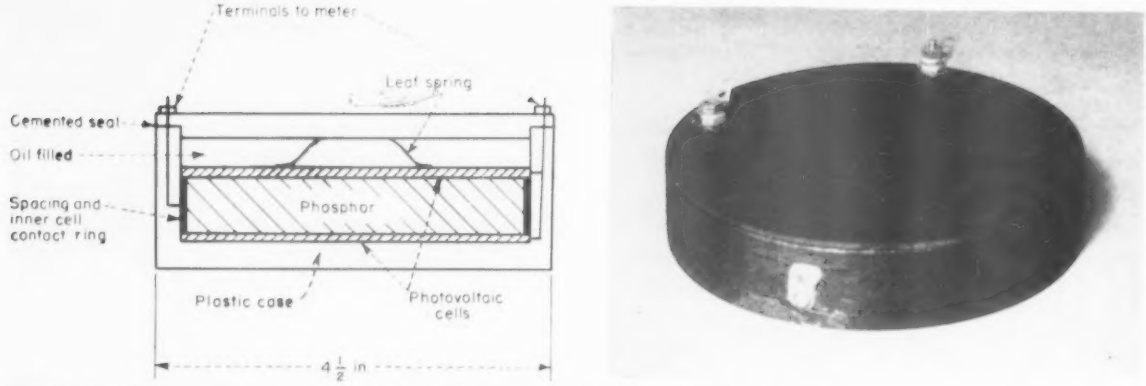

FIG. 2. Sandwich arrangement of photovoltaic-phosphor detector and its cross-sectional diagram

by the filtering action of the second cell, which is interposed between the X-ray source and the phosphor. The shielding effect more than overcomes the added output of two cells. In the second case, the aluminum foil increased the current output by 25%.

If we are interested in the measurement of X-rays, and we are not concerned with the directional response of the detector, then the construction should follow the method of the first or second case. A detector using this configuration would obviously be very directional in its detection properties, but in many instances would be useful because of its greater sensitivity.

Directional Response

The directional properties of the sandwich detector for X-rays are shown in Fig. 4. The ratio of maximum to minimum response is approximately 4:1 (*5*).

As expected, the maximum response occurs when the incident X-radiation is perpendicular to the plane of the phosphor and cells. This may be verified by calculating the order of magnitude of such a directional dependence in a field of 100-kev X-rays ($\lambda = 0.124$ Å).

In the first case, when the direction of these photons is perpendicular to the plane of the detector, they must pass through about $\frac{1}{16}$ in. (0.16 cm) of steel, representing the P.V. cell. For this energy, the mass absorption coefficient, μ/ρ, of steel is approximately 0.4 cm²/gm. The incident field is then attenuated to some fraction of its original intensity

$$I/I_0 = e^{-\mu x} = 0.60$$

This radiation must now pass through a phosphor having an area of 80 cm² and a thickness of about 1 cm. The phosphor contains 15% sodium and 85% iodine. The mass absorption coefficients of Na and I are roughly 0.16 cm²/gm and 5.0 cm²/gm for this energy photons. A weighted average for NaI is then $\mu/\rho = 4.5$ cm²/gm, giving a linear absorption coefficient $\mu = 17$ cm⁻¹. The thickness of a layer of NaI necessary to absorb 90% of the incident X-rays is

$$I/I_0 = 0.1 = e^{-\mu x}$$
$$x = 0.135 \text{ cm}$$

Approximately, then, all of the radiation which filters through the P.V. cell is absorbed in the phosphor.

In the second case, radiation incident in the plane of the detector sees a projected area of phosphor equal to about 10 cm². All of this radiation is absorbed.

We can then expect that the ratio of the amount of radiation intercepted by the phosphor in the first case to that in the second case is roughly

$$R = \frac{\text{Plane}}{\text{Edge}} = \frac{I_0(0.60)(0.100)(80)}{I_0(1.00)(1.00)(10)} = 4.8$$

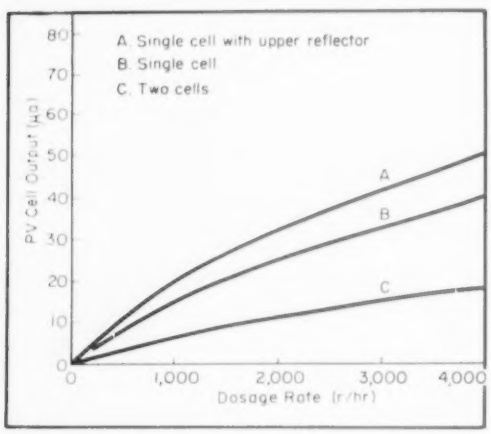

FIG. 3. P.V.-cell response to 140-kv X-rays

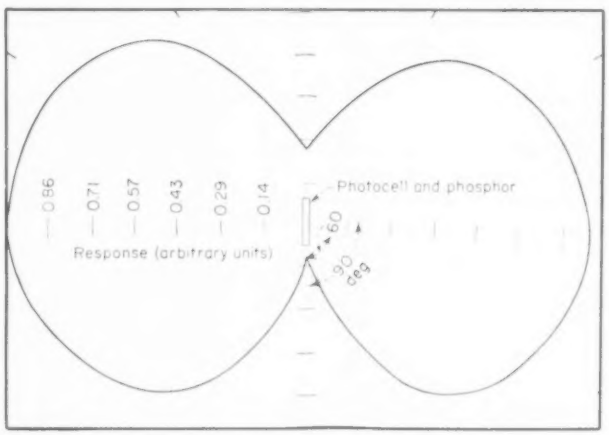

FIG. 4. Directional dependence of sandwich detector

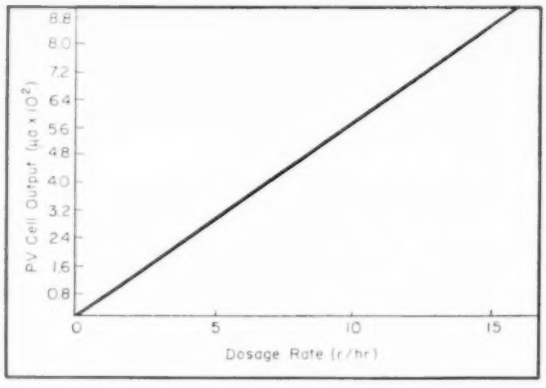

FIG. 5. Response of sandwich arrangement to radium gamma rays

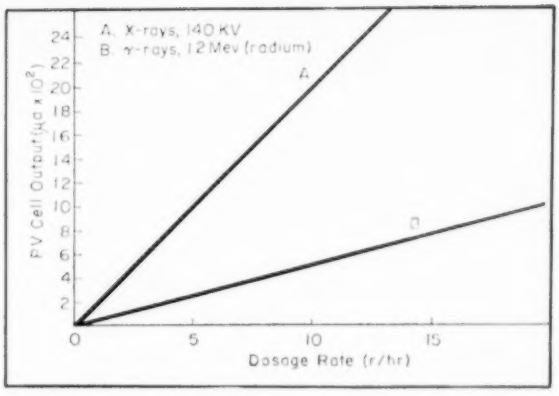

FIG. 6. Energy dependence of detector with X-rays and radium gamma rays

Since the observed ratio was 4:1 under similar conditions, we are not far wrong in our expectations. The optimum geometry to reduce this directional dependence to a minimum would be a sphere of phosphor surrounded entirely by a uniform absorber.

With gamma rays, the detector responds as shown in Fig. 5. The best configuration for detection of gamma rays was the sandwich arrangement, since the cell represents but small shielding to gammas of this energy. Energy dependence of the detector may be determined from these curves. For unfiltered X-rays (140 kv), the sensitivity is about 0.02 μa/r/hr; for gamma rays (radium), it is about 0.0056 μa/r/hr.

From Fig. 6, at a dosage rate of 10 r/hr of both X-rays and gamma rays of the above-mentioned energies, the current output is 20×10^{-2} μa and 5×10^{-2} μa, respectively. Thus, the ratio of the detector output with X-rays to that with gamma rays is approximately 4:1, under these test conditions. Precise measurements on energy dependence have not yet been made, nor has any attempt yet been made to improve this dependence.

Measurements have been made at higher gamma-ray intensities. The version of the detector used in these measurements, taken at Brookhaven National Laboratories, measured 2 in. × $1\frac{1}{4}$ in. × 1 in. The table summarizes these measurements for one tantalum and two cobalt sources (*6*).

P.V. Response to Intense Gamma Radiation

Source type	*Detector reading (μa)*	*Dosage rate (r/hr)*
Ta	43	170,000
Co^{60}	50.5	300,000
Co^{60}	50.5	300,000

Efficiency

The maximum efficiency of such a detector can be calculated by first noting the energy absorbed in the crystal from a given flux of radium gamma rays, then finding the energy of the emitted light and the corresponding cell current. This figure can be compared with the detector's actual performance.

In air, a roentgen of radiation dissipates about 83 ergs/gm. This quantity is approximately the same for all substances in the Mev range, including NaI(Tl). In the tests illustrated in Fig. 5, about 175 gm of phosphor were used. Therefore, for a gamma dosage rate of 1 r/hr at the phosphor, the dissipation amounts to 4 ergs/sec or 4×10^{-7} watts. The maximum theoretical efficiency of the phosphor is about 30% (*4*). Let us say that our crystals are 25% efficient. This means that 25% of the absorbed energy, or 10^{-7} watts, will be reemitted as light. If it is assumed that no light is absorbed in the crystal, and that the detector has 100% geometry, then 10^{-7} watt of light reaches the P.V. cells.

We will make the further assumption that the total emission of NaI(Tl) takes place at a wave length of 4,100 Å. The emission really has some finite band width, in addition to some other smaller emission peaks. From Fig. 9, about 130 μa/mw of irradiation are expected at 4,100 Å. Under these assumptions of nearly optimum conditions, a current of 1.3×10^{-2} μa should result for a radiation dosage rate of 1 r/hr. Under similar circumstances, we measured a current of 0.56×10^{-2} μa (Fig. 5), which compares favorably with the calculated optimum figure.

Conclusions

This instrument has been used to detect X-rays and gamma rays with the recording meter located 2,000 ft from the detector with negligible loss of information.

The detector seems ideally suited to use as a beam location instrument for particle accelerators, for a detection unit on a general purpose X-ray thickness gage, for measurements on high-intensity gamma sources, and for monitoring any system involving gamma radiation levels of 10 r/hr or higher. It can do these things with greater ease and economy than other detectors.

Any increase in sensitivity will probably have to come about by an increase in detector volume, and also by more efficient optical coupling. These improvements cannot, however, be expected to produce too great an increase in sensitivity.

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Properties of Beryllium

Because of its low neutron-capture and high neutron-scatter cross sections, beryllium has attracted great interest as a reactor building material. The engineering possibilities of beryllium are indicated in this survey of its atomic and nuclear, physical, chemical and mechanical properties

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Battelle Memorial Institute, Columbus, Ohio

ENGINEERS AND METALLURGISTS who have the responsibility of designing and constructing nuclear reactors are interested in beryllium primarily because it has a low neutron-capture cross section and a relatively high neutron-scattering cross section.

This compilation and appraisal of atomic and nuclear, physical, chemical, and mechanical data utilizes the best values where feasible; in other cases, values determined by several investigators have been averaged. Where published results appeared dubious, they were excluded from consideration.

Since efforts to produce purer beryllium and to purify beryllium produced by present methods are in progress, new material may appear for which some of the properties listed may no longer be representative. Such an occurrence would require a new determination of the properties in question.

Production of Beryllium . . .

Beryllium is estimated to constitute approximately 0.001% of the earth's crust. The only mineral of present commercial importance is beryl ($Be_3Al_2Si_6O_{18}$). Commercial grades contain 11 to 13% BeO. Beryl ore has never been found in quantities extensive enough to permit mining it for its own sake. Hence, it is produced only as the byproduct of other mining operations.

Brazil, Argentina, and India are now the largest producers of beryl ore, while the principal sources in the United States are in Colorado, South Dakota, and New England. A number of deposits have been found in Canada.

Beryllium products are being produced by three companies in the United States: Clifton Products, Inc., Painesville, Ohio; Beryllium Corporation, Reading, Pennsylvania; and Brush Beryllium Corporation, Cleveland, Ohio. Only the latter two produce the metal.

Production of the oxide from the ore is the first step in producing beryllium metal. The beryl is treated with sulfuric acid after fusion, either by itself or with sodium carbonate or lime to make it more reactive. Another method is fusion with Na_2FeF_6. Iron, aluminum, and other impurities are removed in a series of purification steps. The end product is beryllium hydroxide which can be ignited to the oxide.

The oxide is mixed with carbon and treated at 1,000° C with chlorine. The chloride, fused with NaCl, is electrolyzed at 730° C to produce the metal in flake form, which can be briquetted and melted if massive metal is desired.

In another method, beryllium fluoride, produced by ammonium bifluoride treatment of the oxide, is reduced by magnesium. The reaction takes place at 900° C. When the reaction is completed, the temperature is raised to melt the beryllium which floats to the surface.

Prices for beryllium in 1952 ranged from $50.00 to $200.00 per pound, depending on the form and grade. Billets were available at $90.00 to $125.00 per pound.

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Atomic and Nuclear Properties . . .

The atomic and nuclear properties of beryllium are given in Table 1 and Figs. 1 and 2.

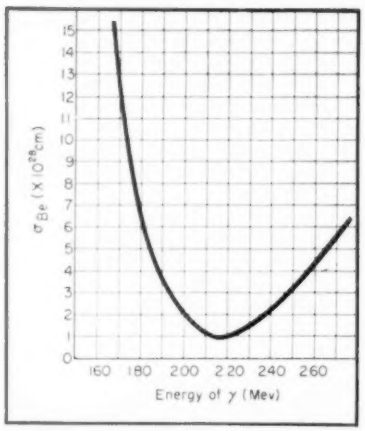

FIG. 1. Photodisintegration cross section for Be^9

TABLE 1—Atomic and Nuclear Properties of Beryllium

Atomic number	*Atomic weight*	*Electron configuration*
4	9.013	$(1s)^2(2s)^2$

Isotope	*Mass*	*Packing fraction*	*Binding energy (Mev)*
Be^6	6.0219	0.00365	4.41
Be^7	7.01916	0.00274	5.33
Be^8	8.00785	0.00098	7.02
Be^9	9.01503	0.00167	6.42
Be^{10}	10.01677	0.00168	6.45
Be^{11}	11.0277	0.00252	5.69

Electron orbit	*Ionization potential (Rydbergs)*	*Orbital radius (Å)*
(1s)	9.3	0.143
(2s)	0.69	1.19

Isotope	*Energy of radiation (Mev)*	*Half life*	*Method of production*
Be^7	0.453-0.485γ	54.5 d	Li^6 (d,n) Li^7 (p,n) B^{10} (p,α) B^{10} $(d,\alpha n)$
Be^8	0.055γ	10^{-15}-10^{-17} s	Be^9 (γ,n)
Be^9	Stable	—	Occurs in nature
Be^{10}	0.56-0.65β, no γ	2.7×10^6 y	Be^9 (α,p) Be^9 (n,γ) B^{10} (n,p) C^{13} (n,d)

Photodisintegration cross section	*Neutron cross section*
See Fig. 1	See Fig. 2

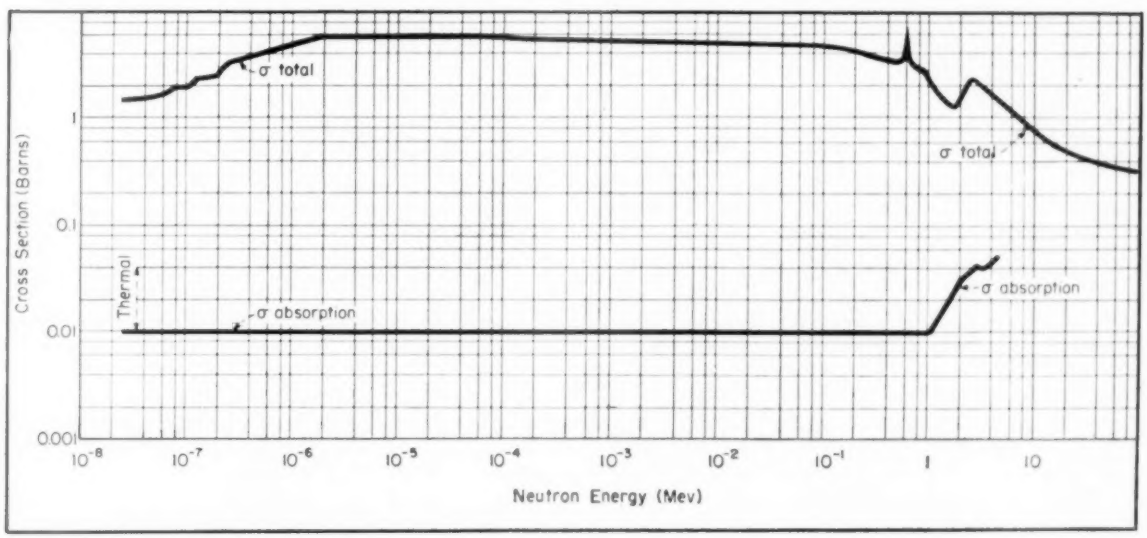

FIG. 2. Neutron cross sections for beryllium

Physical Properties . . .

Table 3 lists selected values of some of the physical properties of metallic beryllium. Reference is given to Tables 2 and 4 for more complete thermodynamic data, to Fig. 3 for coefficients of thermal expansion, to Fig. 4 for thermal conductivity, to Fig. 5 for electrical resistivity, and to Fig. 6 for thermoelectric power. Crystallographic properties of beryllium are given in Table 5.

TABLE 2—Low Temperature Heat Capacity

Temperature (°K)	C_p
10*	0.00
25	0.01
50	0.05
100	0.41
150	1.26
200	2.38

* There is a slight anomaly at 12° K.

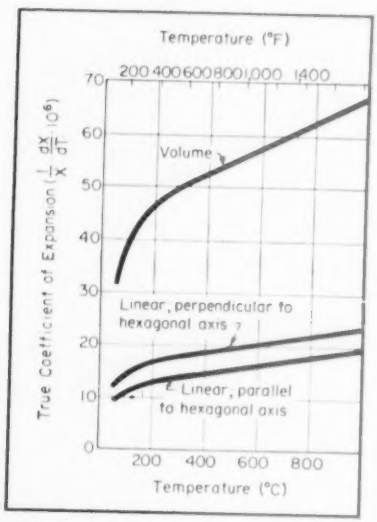

FIG. 3. Coefficients of thermal expansion

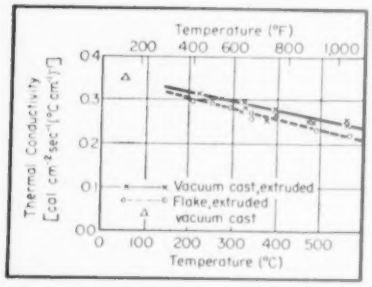

FIG. 4. Thermal conductivity

TABLE 3—Physical Properties of Beryllium

Mass properties		
Atomic weight	9.013 ± 0.0004	(O = 16)
Atomic volume, 25° C	4.877 cm³/mole	
Density, 25° C, X-ray	1.8477 ± 0.0007	g/cm³
Density, 1,000° C, X-ray	1.756	g/cm³
Density, 25° C, 40,000 kg/cm²	1.912	g/cm³
Density, 25° C, 100,000 kg/cm²	1.955	g/cm³
Thermal properties		
Melting point	1,315 ± 20° C	
Boiling point (extrapolated)	2,970° C	
Vapor pressure, solid	905° C	10^{-8} atm
Vapor pressure, solid	1,235° C	10^{-5} "
Vapor pressure, liquid	1,950° C	10^{-2} "
Vapor pressure, liquid	2,970° C	1 "
Heat of fusion	2,335 cal/mole	
Heat of vaporization	53,490 cal/mole	
Heat of sublimation (900–1,280° C)	76,560 cal/mole	
Other thermodynamic properties	See Table 4	
Allotropic transformation	None	
Coefficients of thermal expansion	See Fig. 3	
Contraction on solidification	3%	
Thermal conductivity	See Fig. 4	
Electrical properties		
Electrical resistivity	See Fig. 5	
Superconductivity	Possibly below 11° K	
Electrochemical equivalent	0.04674 mg/coulomb	
Electrolytic solution potential, Be/Be^{++}	E_0 = 1.9 volts	
Thermoelectric power	See Fig. 6	
Optical properties		
Color	Steel gray	
Reflectivity	52–55% (white light), >55% (ultraviolet)	
Emissivity Coefficients		
Solid 6,500 Å	0.61 (to mp)	
Solid 5,500 "	0.61 (to mp)	
Liquid 6,500 "	0.61 (mp to 2,000° C)	
Liquid 5,500 "	0.81 (mp to 2,000° C)	
Magnetic properties		
Mass Susceptibility, −180° C	−0.72 cgs electromagnetic units	
Mass Susceptibility, +20° C	−1.00 "	
Mass Susceptibility, 300° C	−1.20 "	
Hall Coefficient	0.0024 ± 0.0001 cgs electromagnetic units	
Velocity of sound	12,600 meters/second	

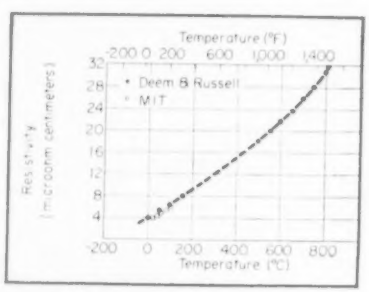

FIG. 5. Temperature vs resistivity

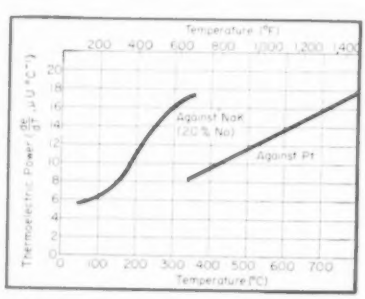
Temperature (°F)
200 400 600 800 1,000 1,200 1,400
Thermoelectric Power ($\frac{dE}{dT}$ μV/°C)
Against NaK (20% Na)
Against Pt
0 100 200 300 400 500 600 700
Temperature (°C)

FIG. 6. Thermoelectric power

TABLE 4—Thermodynamic Properties of Beryllium

Temperature (°K)	Heat capacity,* C_p (cal/mole/°K)		Enthalpy, $H_t - H_0°$, (cal/mole)		Entropy, S_t, (cal/mole/°K)		Free-energy function $-[(F_0 - H_0)/T]$(cal/mole/°K)	
	Condensed	Vapor	Condensed	Vapor	Condensed	Vapor	Condensed	Vapor
298.16	4.26	4.97	465	1,481	2.28 ± .02	32.56 ± .01	0.72	27.60
400	4.56	—	880	—	3.48	—	1.28	—
500	4.85	4.97	1,365	2,484	4.56	35.13	1.83	30.16
600	5.14	—	1,880	—	5.50	—	2.42	—
700	5.43	—	2,430	—	6.34	—	2.87	—
800	5.72	—	3,000	—	7.10	—	3.35	—
900	6.01	—	3,600	—	7.81	—	3.81	—
1,000	6.30	4.97	4,210	4,971	8.45	38.57	4.24	33.61
1,100	6.59	—	4,830	—	9.04	—	4.65	34.08
1,200	6.88	—	5,455	—	9.59	—	5.04	34.50
1,300	7.17	—	6,080	—	10.09	—	5.41	34.90
1,400	7.46	—	6,740	—	10.56	—	5.75	35.27
1,500	7.75	4.97	7,465	7,451	11.05	40.59	6.07	35.62
1,588 (solid)	8.00	—	8,080	—	11.40	—	6.31	—
1,588 (liquid)	5.27	—	11,415	—	12.87	—	6.31	—
2,000	5.27	4.97	13,565	9,931	13.98	42.02	7.20	37.05

* See Table 2.

Chemical Properties . . .

Beryllium has an atomic weight of 9.013 ± 0.0004. Its only recognized valence is +2.

While hydrogen does not seem to react directly with beryllium, it has been shown that hydrogen, probably from water, can cause unsoundness in cast beryllium. Schlessinger, Schaeffer, et al., have prepared a beryllium hydride, but not uncontaminated by the ether solvent in which it was made. Spectrographic lines resulting from BeH_2 and BeH^+ have also been reported when beryllium electrodes were sparked in a hydrogen atmosphere.

Solid pieces of beryllium start to react with air at about 700° C. Reaction with oxygen starts at a considerably lower temperature, but may be slowed down by the formation of an oxide skin. This "protective action," even when the beryllium is molten, has led to difficulties in the analysis for carbon, where the sample is burned in a stream of oxygen. When ignited in air, for example by a magnesium ribbon, finely divided beryllium will burn, but the ignition temperature is fairly high. Polished beryllium will tarnish slightly in air at room temperature, but further reaction is prevented by the oxide skin.

Beryllium reacts with nitrogen above 900° C to form Be_3N_2, but the reaction is slow even at 1,100° C. The reaction is much faster at the same or lower temperatures with NH_3. It reacts readily with $(CN)_2$ at 800° C to form the nitride.

Finely divided beryllium "burns" in sulfur vapor, and selenium and tellurium act like sulfur.

Pure beryllium is resistant to hot and cold water.

Beryllium reacts exothermally with carbon, to form Be_2C, at temperatures not far above the melting point. This causes many difficulties when graphite is used as a melting crucible. Carbon is also reported to react slowly with beryllium at temperatures below the melting point. In addition to the carbide, an acetylide, BeC_2, is known.

Beryllium dissolves in HCl and H_2SO_4 of all concentrations. It is, however, resistant to cold concentrated nitric acid. It reacts with concentrated alkalies and with warm dilute alkalies. It is not attacked by ammonium hydroxide.

Beryllium ignites on gentle heating with the halides. The ignition temperature in iodine is higher than in the other halides. Beryllium also reacts with phosphorus vapor.

Organometallic compounds of beryllium can be produced from beryllium chlorides and the appropriate Grignard reagent.

Molten beryllium reacts more or less completely with most of the other metal oxides. Possible exceptions are ZrO_2 and MgO.

Beryllium is reported to react with the fused alkalies, liberating the alkali metal until an equilibrium is established, but not to react with the fused halides of the alkaline earths or magnesium. It is reported, however, to reduce aluminum halides and halides of the heavy elements.

TABLE 5—Structural Properties

Crystal structure	Close-packed hexagonal		
Space group	D_{6h}^4		
Lattice constants	*a*, kX	*c*, kX	*c/a*
Room temp	2.281	3.577	1.568
200° C	2.287	3.584	1.567
400° C	2.296	3.594	1.566
600° C	2.305	3.606	1.564
800° C	2.315	3.618	1.563
1,000° C	2.325	3.632	1.562

Interatomic distances

d_1	2.221 kX
d_2	2.281 kX

Coordination number 12
Atomic radius 1.123 kX

Mechanical Properties . . .

Typical mechanical properties of beryllium are given in Table 6 and Figs. 7–12. Of particular note are the high elastic modulus, the low Poisson's ratio, and the relatively high ductility in torsion.

As seen in Figs. 8 and 9, there are two temperatures of maximum ductility in tension: 400° and 800° C.

In contrast to the powder-metallurgy products and to cast products, extruded beryllium shows directional differences in tensile properties. Two structural characteristics may account for this effect, the gross longitudinal grain fibering introduced by the extrusion process and the preferred orientation of the grains. Gross fibering is virtually eliminated by annealing at temperatures above 750° C, but preferred orientation remains, even up to 1,000° C. Since annealing does not appreciably improve the transverse tensile properties, it would appear that directional properties are largely associated with preferred orientation.

Extrusion temperature and surface condition have an appreciable effect on the room-temperature tensile properties. In the as-extruded and machined condition, the ultimate strength dropped off rapidly as the extrusion temperature was increased, and the per cent elongation was only a few tenths of a per cent for all extrusion temperatures.

If the surface was removed by etching, the ultimate tensile strength was increased for extrusion temperatures of 900° C and above, and the elongation was markedly improved. The surface removal had little effect on properties of metal extruded at 500° or 700° C. The microstructure of the as-extruded material showed that extrusion at 500° and 700° C left the metal in a completely cold-worked condition. Partial recrystallization was evident in the metal extruded at 900° C and complete recrystallization in metal extruded at higher temperatures. If the metal is in a severely cold-worked condition throughout, it is inherently strong and brittle, and surface damage from machining, if present, has no appreciable effect on properties.

The effects of surface damage can be completely eliminated by annealing for one hour at 800° C.

The effects of annealing on mechanical properties at room temperature can be explained by recrystallization at temperatures above 750° C and lack of recrystallization at lower temperatures.

The equicohesive temperature for beryllium at ordinary strain rates seems to be about 450° C.

TABLE 6—Mechanical Properties of Beryllium

Modulus of elasticity	38.8 to 42.6 × 10^6 psi
Shear modulus	19 × 10^6 psi
Poisson's ratio	0.02
Tensile properties	
Extruded	See Figs. 7, 8, 9
Cold pressed and sintered ("QT"), annealed 800° C	34,100 psi UTS, 1.9% elongation in 2 in.
Hot pressed (KAPL), annealed 800° C	33,000 psi YS (0.2% offset); 46,000 psi UTS; 1.15% elongation in 2 in.
Stress rupture properties	See Fig. 10
Compressive properties	
Cast and forged, 9,200 psi proportional limit, 35,300 psi YS (0.2% offset)	
Hardness (room temperature)	
Commercial vacuum cast	107–125 Bhn
Commercial extruded	116–121 Bhn
Brush "Q"	119–150 Bhn
Extruded vacuum cast	106 Bhn
Extruded turnings	112 Bhn
Extruded flake	132 Bhn
Hardness (elevated temperature)	
300° C	88 Bhn
400° C	90 Bhn
500° C	74 Bhn
600° C	61 Bhn
700° C	50 Bhn
800° C	21 Bhn
900° C	11 Bhn
1,000° C	9 Bhn

Unnotched impact strength (ft-lbs)	*Extruded*	*"QT"*
Room temp	0.12	0.12
100° C	0.13	0.13
200° C	0.25	0.18
300° C	0.26	0.22
400° C	1.21	0.19
500° C	0.22	0.30
600° C	1.36	0.19
700° C	>2	0.31
800° C	>2	0.28
Torsional properties	See Figs. 11 and 12.	

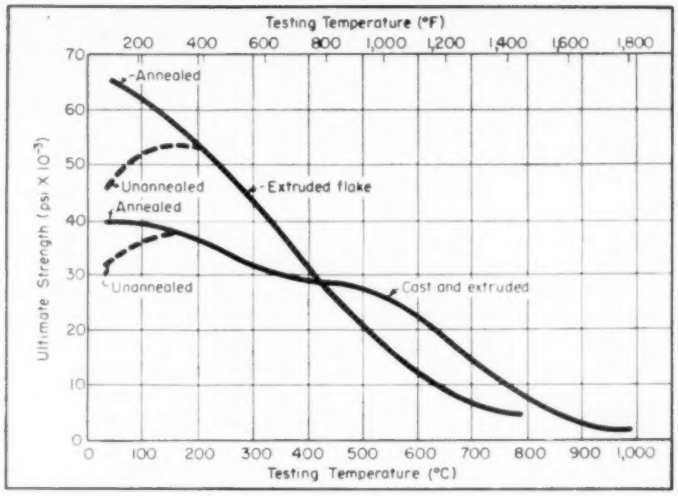

FIG. 7. Ultimate strength of Be vs testing temperature in short-time high-temperature tensile tests

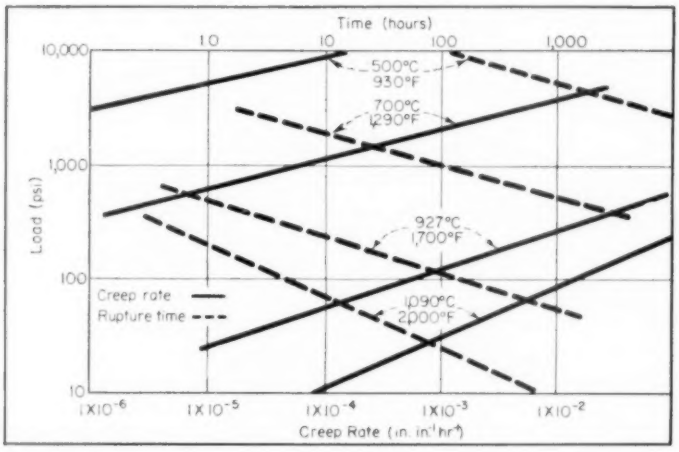

FIG. 10. Stress rupture and creep data for Be

TABLE 7 Corrosion of Be by Ga

Temperature (° C)	*Results*
800	No attack on Be
600	Diffusion of gallium into Be
500	No attack, no diffusion

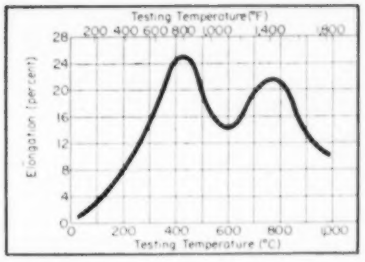

FIG. 8. Elongation vs temperature

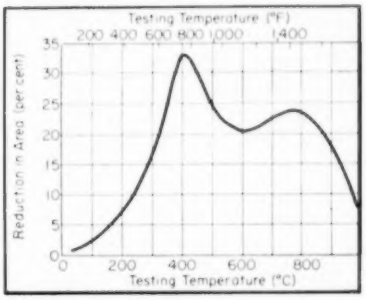

FIG. 9. Variation of reduction in area

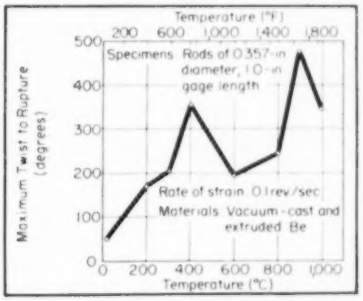
Temperature (°F)
Specimens: Rods of 0.357-in diameter, 1.0-in gage length
Rate of strain: 0.1 rev/sec
Materials: Vacuum-cast and extruded Be
Maximum Twist to Rupture (degrees)
Temperature (°C)

FIG. 11. Effect of temperature on maximum twist to rupture

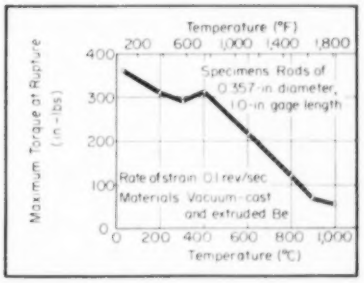
Temperature (°F)
Specimens: Rods of 0.357-in diameter, 1.0-in gage length
Rate of strain: 0.1 rev/sec
Materials: Vacuum-cast and extruded Be
Maximum Torque at Rupture (in.-lbs)
Temperature (°C)

FIG. 12. Effect of temperature on maximum torque at rupture

Corrosion . . .

Beryllium corrosion has been studied in air, water, and liquid metals.

Corrosion in air. There is no severe corrosion of beryllium in air at 400° C in 200 hr. The times for onset of noticeably severe corrosion are reported as 60 hr at 700° C, 12 hr at 800° C, and 1 hr at 900° C. These tests were made on extruded, vacuum-cast material. The oxidation is a grain-boundary attack. The oxide formed is hard and brittle and is difficult to retain on metallographic polishing.

Howe, at Knolls Atomic Power Laboratory, found that, with extruded beryllium, 1 hr at 700° C gives a scarcely perceptible color change, 1 hr at 800° C gives a white film, and 1 hr at 1,000° C gives a heavy white scale. Electron-diffraction studies showed oxide on all three specimens, but no nitride. As a check, no oxide was

found by this method on the as-machined specimens. However, analysis of the scale formed at 1,000° C showed 0.75% Be_3N_2. The same author reports that 72 hr at 700° C produced a very thin, spotty white film. There was an evolution of unidentified volatile matter from below the surface. He also states that the resistance of beryllium to oxidation at 700° C seems to be greater than that of 18-8 stainless steel.

Gordon, at Massachusetts Institute of Technology, found that Process "Q" metal is inferior to extruded vacuum-cast material in corrosion resistance to 700° C and 800° C air. Extruded flake material is also inferior to extruded vacuum-cast material at these temperatures.

Corrosion in water. Localized impurities may be the cause of corrosion. Occasional samples have been quite resistant to 300° C water.

Massachusetts Institute of Technology found that Process "Q" and extruded flake beryllium are inferior to extruded vacuum-cast metal in corrosion resistance in boiling water.

The U. S. Naval Experiment Station subjected cast beryllium to static tests in distilled water (pH 6.0 to 6.5) at 260° C under an oxygen pressure of 700 psi. They found that the samples exhibited small nodules and in some instances sharp barbs after exposures of 20 hr to 30 days. The results suggest that corrosion pitting should be considered.

In a dynamic test for 19½ days in oxygen-saturated water and with a surface velocity of 27 ft/sec, severe intergranular corrosion, with actual decomposition of the sample, took place. One sample was broken through in 210 hr and had to be replaced.

Corrosion in liquid metals. Argonne National Laboratory determined that the alloying of gallium with beryllium at 600° C is dependent on the surface condition of the beryllium. The results obtained when gallium was first saturated with beryllium (about 1%) are given in Table 7.

At 600° C, fine-grained beryllium was more readily attacked by gallium than was coarse-grained metal, indicating, perhaps, grain-boundary attack as the mechanism of the corrosion.

The National Physical Laboratory showed beryllium specimens were not affected after 24 hr at 700° C in liquid bismuth, lead, or tin.

Stress corrosion. No evidences have been found of stress-induced corrosion in beryllium.

Fabrication of Beryllium . . .

Methods for melting beryllium in a vacuum and in air under an argon blanket have been developed. The disadvantages of various crucible materials are listed in Table 8.

Graphite is the best all-around mold material, and National Carbon Company's "CS" grade has been most satisfactory. Steel coated with beryllia or alumina, and alumina coated with beryllia, have also been used. Magnesia or zirconite sand molds have been used with some success.

Both static and centrifugal casting of beryllium have been successful. A low pouring temperature is recommended for all types of casting.

The following casting difficulties have been encountered with beryllium:

1. Unsoundness caused by absorption of hydrogen from water vapor and subsequent release on solidification. Thus, everything concerned with melting should be dried before use.

2. Unsoundness caused by carbon-oxygen reaction during solidification.

3. Folds, cold shuts, etc., caused by entrapped dross. Turbulence should be avoided while pouring.

4. Hot shortness, caused by low strength at a temperature near the solidus. Normal shrinkage should not be restricted by the mold. Early stripping from the mold has some advantage.

5. Cold brittleness may cause cracking.

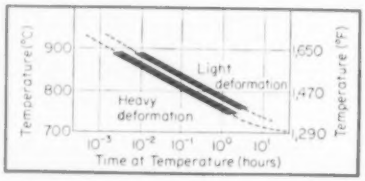

FIG. 13. Time-temperature relationship for recrystallization of Be

6. Large grain size, which produces poor properties and poor machinability. This may be alleviated to some extent by rapid cooling, but this, in turn, promotes cracking. Alloying with small amounts of aluminum is beneficial.

Brush Process "Q" metal is a powder-metallurgy product. It is made by cold pressing beryllium into the desired form and then sintering. Powders can also be made into massive form by hot pressing at 750 to 1,000 psi in a graphite mold at 1,100° C.

It is possible to hot extrude beryllium in bars, tubes, ribbons, hexagons, and plates up to 4½ × 1½ in. The preferred material for extrusion has been vacuum-cast billets, but flake and powdered beryllium have also been extruded into useful shapes.

The most useful technique has been to "can" the billet in a mild steel can and to copper plate the can for lubrication. The customary extrusion temperature is approximately 1,050° C. Extrusion is possible at lower temperatures, but power consumption is higher.

Beryllium cannot be rolled at room temperature, but rolling at temperatures between 350° and 800° C leaves the material in a cold-worked condition. Rolling at higher temperatures gives a hot-worked product. Best practice for hot rolling is to roll just above the recrystallization temperature.

TABLE 8—Crucible Materials for Melting Beryllium

Material	*Disadvantages*
Beryllia	Scarcity of large sizes; high cost; short life of "commercial" crucibles; attacked by fluoride flux
Magnesia	Must be high purity; commercial magnesia crucibles not satisfactory because of contamination; attacked by fluoride flux
Graphite	Exothermic reaction starts within 100° C above melting point
Graphite-bonded silicon carbide ("Tercod")	Satisfactory in slightly oxidizing atmosphere, but not in vacuum; success depends on formation of oxide coating

Foil, ½ to 2 mils thick, has been rolled from a 4¼-in.-diameter cast billet.

The forging characteristics of beryllium are similar to those of magnesium. It forges well, in closed dies, at 400° C. Graphited oil or grease is a suitable lubricant.

Heat treatment. The major effect in the heat treatment of beryllium is recrystallization and grain growth. Recrystallization temperature varies with the degree of prior deformation, as illustrated in Fig. 13. The rate of grain growth increases rapidly as the temperature is increased above the recrystallization temperature.

Preferred orientation introduced by extrusion or rolling persists up to 1,000° C, although gross fibering may be eliminated by recrystallization at temperatures as low as 725° C.

Joining. Beryllium can be pressure welded in vacuum or argon at 1,200° C. Beryllium can be welded to itself and to aluminum without filler rods, using straight-polarity, helium-shielded arcs. Beryllium can be brazed, using aluminum-filler rod, or by using an aluminum-foil technique. "Phos-Copper" has also been used to braze beryllium.

Beryllium can be successfully soldered by painting the surface with a paste of Cu_2Cl_2 in turpentine, heating to produce a copper deposit, and then soldering in the normal manner.

Machinability. Beryllium is a difficultly machinable metal, and under certain circumstances machining has an adverse effect on tensile properties.

Extruded and rolled beryllium is less sensitive to grinding cracks than beryllium that has only been extruded, which is, in turn, less sensitive than cast beryllium. To minimize difficulty with grinding cracks, the Norton Company developed a special abrasive wheel, No. 57A46-G12VBEP No. 24, treated.

In tool-wear studies, "Q" metal was superior to other types of beryllium tested. Oriented, extruded metal gave less tool wear than cast metal, when tested in the direction of extrusion. Transverse to this direction, it showed no superiority. Heat treatments removed this directionality.

Applications of Beryllium . . .

Beryllium in sheet form is used as window material in X-ray tubes. Because of its low atomic weight, it allows X-rays to pass through 17 times better than an equivalent thickness of aluminum which was previously used for these windows.

Beryllium is used in small amounts to reduce the oxidation of magnesium and magnesium alloys during melting and casting. It can also be used (about 0.01%) as a deoxidizer in place of phosphorus in the production of copper castings. The product has a higher density and a greater electrical conductivity than if phosphorus were used.

In aluminum-base alloys, 0.005% beryllium restrains oxidation in the molten state and refines the grain. Similar quantities added to magnesium-base alloys were found to increase corrosion resistance.

The largest prewar use of beryllium was in beryllium-copper alloys. These precipitation-hardenable alloys, containing about 2% beryllium or less, have wide applications where high-strength, nonsparking, nonmagnetic parts or tools are desired. Master beryllium-copper alloys, containing about 4% beryllium, are produced by reduction of the oxide with carbon in the presence of copper or copper oxide.

Because of its high reflectivity, particularly in the ultraviolet range, beryllium is used as both oscillating and stationary mirrors.

Beryllium has been used in conjunction with an alpha-ray emitter such as radium as a source of neutrons.

Possible uses. Beryllium has been proposed and is being considered as a moderating material for thermal-energy nuclear reactors. This interest stems from its combination of low atomic weight, low thermal-neutron-capture cross section, and relatively high scattering cross section. For this same reason it is also of interest as a reflecting material for nuclear reactors. It is also receiving attention as an alloying element and as a cladding material for fuel elements.

In these applications, the metal and the oxide are receiving the most attention, although other materials such as the carbide and the nitride are being considered.

A minor proposed use is as a possible scavenger for other metals. As a deoxidizer for sodium, beryllium was not too successful.

Precautions in use. Beryllium and its compounds present potentially serious health hazards. Consequently, competent medical authorities should be consulted in establishing precautions for plant and laboratory workers. Once established, these regulations should be observed strictly by the personnel. Any interpretation of regulations should be approved by medical authorities.

* * *

This paper is based on a survey of data available before April, 1949, performed under Navy Department Bureau of Ships, Contract NObs-46822, and Atomic Energy Commission Contract W-7405-eng 92. The survey report is unavailable and declassified.

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Radial Response of Typical End-Window G-M Tubes

By AVRAHAM AMITH* and W. WAYNE MEINKE
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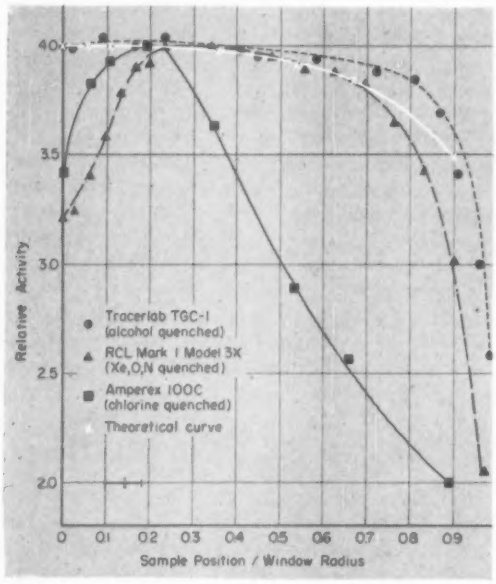

RADIAL SENSITIVITY of typical G-M tubes. Theoretical curve for 100% efficiency is shown in white

In the course of investigating the characteristics of several types of commercially available G-M tubes, we measured the radial response to a collimated beta-particle source of an argon-filled chlorine-quenched "infinite life" tube (Amperex 100C), an argon-alcohol quenched tube (Tracerlab TGC-1), and an argon-filled tube quenched with a small amount of xenon, oxygen and nitrogen (Radiation Counter Laboratory Mark I Mod. 3X).

Several authors (*1, 2*) have reported experiments defining the sensitive volumes of certain early types of G-M counters for beta particles or cosmic rays. Saurer (*3*) has made an extensive study of the radial response to X-rays of argon-filled alcohol-quenched G-M tubes. It is furthermore generally recognized that there is considerable radial dependence of the response in the so-called "infinite life" fillings using chlorine and other halogens as quenching gases. Amperex (*4*) has published a radial response curve for X-rays in their chlorine-quenched tubes, and note the fact that for beta particles the radial response may be 5–10% less in chlorine-quenched than in alcohol-quenched tubes. Others also report using less than 100% efficiency for counting beta particles in chlorine-quenched tubes because of the reduction of sensitive volume in these tubes, but they do not show exactly how the response varies with radius. The work reported here was an attempt to measure this radial response as accurately as possible.

Experimental

A 3.25-cm cylindrical collimator, 1.64 cm in height, was fabricated from 24ST aluminum alloy. A collimating hole 0.13 cm in diameter was drilled in the center of the aluminum cylinder. A strong pure beta-ray source (Sr^{90}–Y^{90}) was secured inside the collar of the collimator to produce a pencil of beta-particles through the collimating hole. The collimator was so adjusted that its top was 1 mm beneath the mica window of the end-window counters. The collimator was held by a clamp which in turn was attached to a movable vernier on a scale giving reproducibility of position to within 0.1 mm. The hole of the collimator was aligned with the exact center of the Geiger tube, and the counting rate was determined at different positions from the center. The sample and collimator were held and the tube mounted in a plastic stand arrangement manufactured by Technical Associates. The stand was mounted on a wood base with no shielding except for a ½-in. lead cylinder around the top of the Geiger tube to reduce the background. With this arrangement beta scattering was minimized.

An important feature of the experiment was the use of a "blank" which was identical with the collimator except that it had no collimating hole. Measurements were taken on the blank to correct for the counting effects of the bremsstrahlung resulting from the interaction of the beta particles with the aluminum collimator. All measurements were at room temperature.

Results

Experimentally determined curves of radial sensitivity for the three tubes are shown in the illustration. Each curve was adjusted so that its maximum occurred at about the same relative activity as the other curves. The curves of the three tubes were made further comparable by using as abscissa the ratio of the sample position (radial distance from the center) to the window radius.

The errors of the experimental

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points are given in the lower left-hand corner of the graph. The experimental width of the collimating hole is represented by the horizontal bracket; probable error of the activity measurements is given by the vertical bracket.

It is possible to calculate from published formulas (*5*) the theoretical radial response curve for 100% counting efficiency, obtained from strictly geometrical factors, for the experimental set-up used. This theoretical curve is identical for all three tubes and is shown in white in the illustration. The curve for the argon-alcohol quenched tube is almost identical with the theoretical curve, but the other tubes show variations from this curve. The effect of the center wire is different in the two "infinite life" tubes.

When the relationship between the radial position of the source and the tube's response was measured as a function of tube voltage, it was found that the shape of the response curve remains unaltered throughout the range of voltages of the tube's plateau. This effect was investigated in detail with the Amperex chlorine-quenched tube, and it was found that in the spontaneous discharge region, the curve is tremendously magnified, the "axial dip" much deeper, and the curve much more pointed. The data on hand, however, do not indicate clearly whether there is any significant change at the different voltages in the ratio of the counts at the center to that at the maximum point of the curve. This ratio seems to remain approximately the same, even at discharge voltages.

When plateau curves were taken at each radial position indicated by a point on the curve for the Amperex 100C tube, the slope of the plateau varied considerably. At the center wire, a plateau slope of 2.5%/100 volts was obtained; progressing towards the tube wall, slopes of 2.5, 2.1, 2.3, 4.9, 6.7, 6.1, and 7.1 were then obtained. Since these values are precise to within a few tenths, the apparent increase in slope as the source approaches the tube wall is real.

This same phenomenon has been observed by Hull (*6*) in the study of cylindrical G-M tubes with the "xenon, oxygen, nitrogen" infinite-life fillings.

Discussion

The primary purpose of this work was to learn about the Amperex chlorine-quenched tube. The other two tubes were examined only casually for a comparison. The work presented here is a survey of only a few of the problems involved in using the infinite-life tubes and will serve as a starting point for further detailed studies.

The main advantage of using chlorine-quenched tubes is their "infinite-life." On the other hand, this study has shown some disadvantages in the use of these tubes that must be considered in the evaluation of data taken with them. Assays of radioactive materials obtained with these tubes will be accurate as long as they are compared to a standard of the same material and of the same size and physical characteristics. In following the decay of samples, especially unevenly distributed samples, or in comparing sample with standard on these tubes, it is of utmost importance to have absolutely reproducible geometry to insure counting with constant tube sensitivity.

One further interesting development of this work shows that the counting rate at the edge of the Geiger tube increases much more rapidly with voltage than the rate at the center. Hence the shifting of the voltage for a count from one point to another on the plateau means that certain portions of a sample (especially an uneven one) will be emphasized more than other portions.

The width of the collimating hole used was limited by the ratio of beta counts coming through the collimator to bremsstrahlung counts. With an aluminum collimator, a hole smaller than 0.13-cm was found to give an undue amount of bremsstrahlung background. This background could be reduced by fabricating the collimator out of a material (such as Lucite) of lower Z than aluminum.

* * *

This work was supported in part by the Atomic Energy Commission. The isotopes used in this work were obtained from the Oak Ridge National Laboratories on authorization from the Isotopes Division, U. S. Atomic Energy Commission.

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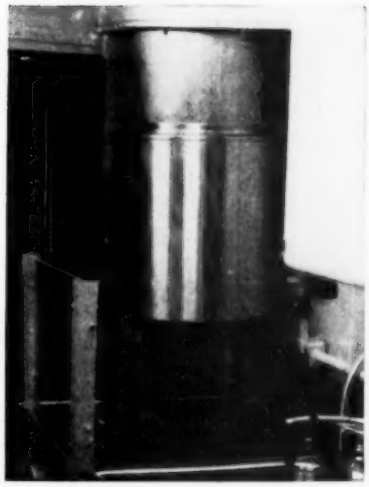

INEXPENSIVE exhaust hood takes little space

Exhaust Hood for Mobile Laboratory

By MORTON I. GOLDMAN
Radiological Health Training Section
Environmental Health Center
U. S. Public Health Service
Cincinnati, Ohio

The exhaust hood installed in the Radiological Health Field Training Unit was designed to provide a maximum of utility at a minimum of cost in the small space available. As shown in the photograph, the basic structure consists of two concentric stainless-steel cylinders. The larger, 14 in. in diameter, is movable along the length of the smaller, 12 in. in diameter. Set screws are used to hold the larger cylinder at any point so that the required height of the hood above the work table can be maintained. The minimum distance between work top and hood is 3 in.

Both cylinders are coated with Tygofilm strip plastic for ease of decontamination. A plastic shield is provided around the work surface.

The exhaust blower is a 10-in. kitchen exhaust fan controlled by a three-speed switch to provide capacities of 500, 750, and 1,000 ft³/min. These exhaust rates will provide "face" velocities of up to 150 ft/min.

The cost of the entire unit, installed, was about $75.00.

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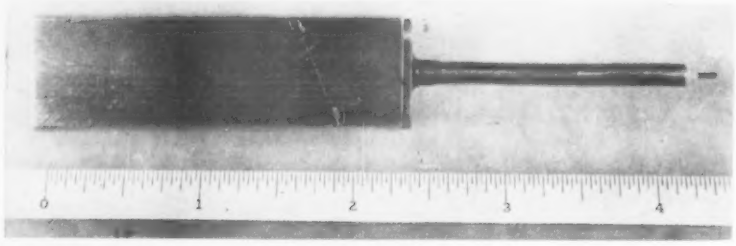

FIG. 1. Assembled thermopile, with connecting coaxial cable

Neutron Flux Measurements with Thermopiles

By ALWYN C. LAPSLEY

Argonne National Laboratory
Chicago, Illinois

A problem often encountered by those who work with nuclear reactors is the one of measuring the neutron flux. A simple, compact and accurate method is desirable. Several techniques have been developed, and each seems to have its attractions and limitations. The method described here uses a well known thermoelectric phenomenon, the Seebeck effect.

The mechanism of this effect is quite simple. If two dissimilar metallic wires are connected at their end points, and if the two end points are at different temperatures, an electric current will flow in their circuit. The magnitude of this current will be a function of the temperature difference.

This principle has been long employed by engineers for measuring temperatures at inaccessible points. A procedure, for example, might be to choose iron and constantan for the dissimilar metals. One of the junctions would be maintained at a fixed temperature in an ice water bath, and the other would be located at the region whose temperature was to be determined. Their electric circuit would then be closed through a voltmeter. By choosing circuit parameters so that the voltmeter had practically all of the circuit resistance, it would indicate essentially the electromotive force developed between the two junctions. This device is a thermocouple.

The thermocouple has one rather serious limitation. For normal temperature differences, the developed voltage is rather small for most pairs of metals. Chromel-constantan thermocouples have about the highest sensitivity with an emf of just over 50 $\mu v/^\circ$ C. To increase this signal, a common procedure is to make up a series of junctions, switching back and forth between any two metals. If alternate junctions are then placed at the two different temperatures, the thermocouple emf of each pair of junctions adds. Such a unit is a thermopile.

Early in the development of reactor

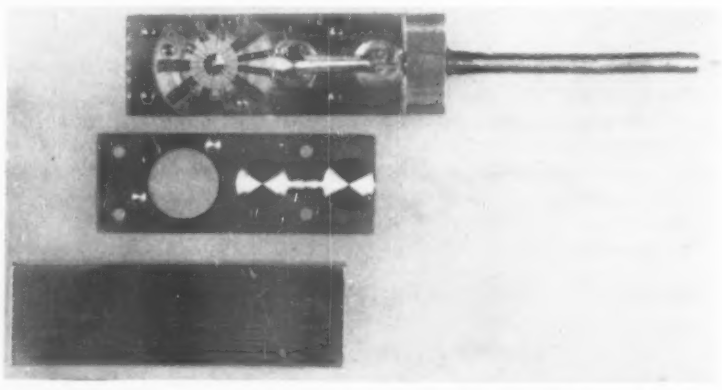

FIG. 2. Component parts of thermopile. Boron coating has not been applied

technology, it was realized that thermopiles might be used conveniently to indicate the neutron flux. A unit was constructed by Robert J. Moon and his associates at what is now the Argonne National Laboratory.* They used alumel-chromel as the thermocouple elements, which were butt welded together to produce a total of 21 pairs of junctions. Alternate junctions were coated with normal boron. As a result of the nuclear reaction

$$_5B^{10} + {}_0n^1 \rightarrow {}_3Li^7 + {}_2He^4 + 2.75 \text{ Mev}$$

these junctions were heated by neutron bombardment. The other junctions would remain essentially at the surrounding temperature. An emf would thus be developed across the entire unit. Its magnitude would be controlled by the temperature of the hot junctions which would in turn be a function of the rate of neutron bombardment, or the flux. Thus, the developed voltage would be a measure of the neutron flux.

It may be noted that this unit differed from those described earlier in that the cold, or control, junctions were not held at a constant temperature. Should the temperature of the unit vary, however, the cold and hot junctions would be affected equally, with no spurious signals introduced. The unit thus automatically compensated for ambient temperature variations.

Recently, a quite compact neutron thermometer operating on the same principle has been developed jointly by the Brown Instruments Division of the Minneapolis-Honeywell Regulator Company and Argonne. The neutron-sensitive element consists of the standard Brown Radiamatic Thermopile onto which an emulsion of B^{10} has been coated. This is packaged into a water-tight 28 aluminum housing. Figure 1 is a photograph of the complete unit. The complete package is made from 0.75-in. diameter stock, but flats on it reduce the thickness to $\frac{5}{16}$ in. Total length is about $2\frac{1}{2}$ in.

The Brown Radiamatic around which this is built is a ten-element thermopile so constructed that the alternate "hot" junctions meet in a small central area. Normally, it responds to light energy which is focused onto this central region. The neutron sensitivity is produced by coating this area with the boron emulsion. A view of the component parts is shown in Fig. 2. Note, however, that the coat-

* G. Barbaras *et al.*, AECD-2975 (1945).

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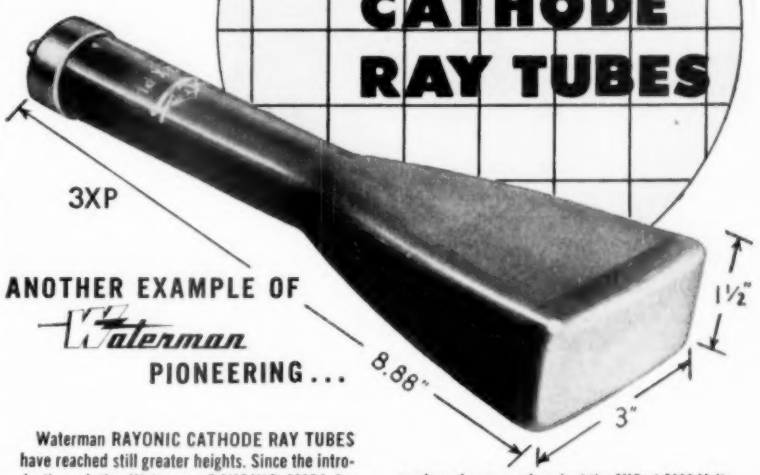

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LENGTH	8.875 inches
BASE	Loctal
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FILAMENT	6.3 Volts
	0.6 Amps.
ANODE # 2	2000 Volts
	Max. 2750 Volts
ANODE # 1	400 to 690 Volts
GRID # 1	−22.5 to −67.5 Volts
DEFLECTION FACTOR IN VOLTS/INCH	
D1 to D2	68 to 92
D3 to D4	25 to 35

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ing has not been applied on this particular unit.

The thermocouple elements for the Radiamatic are chromel and constantan. There are ten junction pairs, and the temperature sensitivity is 0.5 mv/°C. The metal laminae supporting the thermocouple wires are imbedded in mica. The centrally located boron coating (15 mg boron in an emulsion with Sauereisen No. 1 cement) is 3.5 mm in diameter by 1.0 mm thick.

These units have a sensitivity of 1 mv at 10^{11} n/cm²/sec flux. Their output has been found to be linear to at least 3×10^{12} flux, but it is believed the sensitivity would decrease at higher values. Their lower limit of usefulness would be governed by the sensitivity of available instruments.

In very high flux, burnout of the boron would become a factor. It may be noted that the boron has a very high cross-section for neutron capture, and the coating is black to neutrons which strike it. The sensitivity is thus a function of the area the boron presents to the flux, and not the quantity of deposit. Burnout will be controlled by the rate at which this area decreases. Rough calculations indicate that at a 10^{12} flux, the sensitivity would decrease 1% in three years.

These neutrons thermometers follow flux changes quite rapidly. This was noted by an experiment performed with the Argonne heavy-water reactor. The rate of decay of the voltage was observed following a reactor shutdown from a flux of 3×10^{12}. Ten seconds after the shutdown, the signal was 3% of its initial value, and after two minutes, it was down to 0.3%.

ADDENDUM

In later use of his "Method for Presenting Liquid Samples to the Flat Surface of a Scintillation Crystal" (NU, April '53, p. 52), Dr. Sear has found that the polyethylene coils described in the article absorb or react with free iodine that cannot be removed by any type of washing. If any of this free iodine is radioactive, it will be impossible to obtain the original background when the coil is empty.

This is a limit on the use of this method of sample presentation only as long as polyethylene is the coil material. If the coil were made of glass, absorbed iodine could be removed easily by washing with dilute hydrochloric acid.

Concomitant *In Vivo* Measurement of Regional Erythrocyte and Plasma Concentrations Using I^{131} and P^{32}

By ALLEN F. REID and BEN WILSON

Departments of Biophysics and Surgery, Southwestern Medical School University of Texas, Dallas, Texas

The concomitant *in vivo* measurements of the concentrations of erythrocytes and plasma in a prescribed region may be accomplished by double-labeling—labeling the erythrocytes with one appropriate radioisotope and the plasma with another. In this article the method is described in application to the concomitant measurement of erythrocyte and plasma concentrations in a dog's lung. The erythrocytes are tagged with phosphorus-32 which emits strong beta rays and no gamma rays; the plasma is tagged with iodine-131 in iodinated albumin (*1, 2*) which emits gamma rays and weak beta rays. The method may be conveniently described under three categories: (1) labeling of the cells and plasma, (2) measurement of radioactivity, and (3) interpretation of results.

Labeling

About 150 mg bovine serum albumin is labeled with I^{131} according to a method described by Gilmore (*3*) that gives greater than 90% yield. A maximum of 2.5 μg of iodine is used, keeping the iodine-to-albumin mole ratio below 2 to minimize possible antigenicity. No antigenic reactions were detected in any experiments using this preparation.

A 10-ml heparinized venous blood specimen is obtained from the experimental dog at least 2 hr before the experiment is to start. A Krebs-Ringer phosphate buffer solution (5 ml) containing 15 mg of dextrose and 1.2 mc P^{32}/kg dog weight is added to the washed cells; this mixture is incubated for 2 hr at 37° C with agitation every 30 min. After incubation the blood is centrifuged, the supernatant is removed, and the cells are washed twice with phosphate-buffered isotonic saline; the last washing is done not more than 30 min before use.

Aliquots of the labeled cells and albumin are measured for their radioactivity. A 7-ml suspension of cells and albumin is made in phosphate-buffered pyrogen-free isotonic saline which contains about 400 μc I^{131} and 400 μc P^{32} for each kilogram of body weight of the dog. Of this, 5 ml is injected and the remainder retained for control measurements.

Measurement

For the *in vivo* measurements of radioactivity, a 2.7-cm (internal diam-

Calculation of RBC and Plasma Concentration in Dog Lung*

Blood		*In Vivo Measurements*	
Hematocrit (H)	0.475	Direct γ-count (A)	148.5
Count with thin filter (E)	1,619	γ-count with plug (B)	85.2
Count with thick filter (F)	51.6	Direct β-count (C)	148.5
		β-count with filter (D)	117.6
μc I^{131}/cm³ plasma, $6.65 \times 10^{-2}[E/(1-H)]$	6.55	μc I^{131}/cm³ tissue, $3.58 \times 10^{-2}(A-B)$	2.26
μc P^{32}/gm RBC, $1.81 \times 10^{-3}[(E-F)/H]$	5.96	μc P^{32}/gm tissue, $4.9 \times 10^{-2}(C-D)$	1.51
$\frac{\text{cm}^3 \text{ plasma}}{\text{cm}^3 \text{ tissue}} = \frac{2.26}{6.55} = 0.345$		$\frac{\text{gm RBC}}{\text{gm tissue}} = \frac{1.51}{5.96} = 0.254$	

* Counts are rate per minute on scale of 64, corrected.

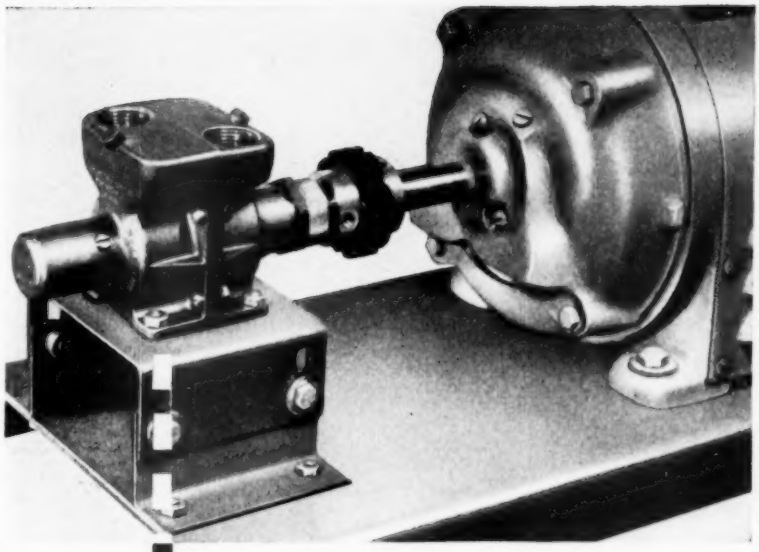
eter) brass tube closed at one end with a 1.5-mm transparent plastic disk is inserted through the bed of a resected rib segment in the chest wall of the dog and sutured in place so that the chest is airtight and the plastic window is contiguous to the lung. Inside this tube is inserted a 3.8-cm lead collimating tube, and a mica end-window G-M counter tube. A 3-cm lead shield surrounds the brass tube and the counter tube for its entire length and is suspended from the ceiling so that respiratory excursion of the dog is not impaired by the weight of the shield.

Two different collimating tubes are used alternately. One is a gamma-ray collimator (*4*) which limits counted activity essentially to gamma rays from a definite volume of the lung and background activity. Provision is made for insertion of a 2-cm-deep lead plug which cuts out most of the gamma rays from the specified lung volume. If A is the counting rate without the plug and B is the counting rate with the plug, then $[A - B]$ is a measure of the gamma rays from the lung volume.

Alternately, a beta-ray collimator is used which admits both P^{32} beta rays and I^{131} gamma rays. Because the range of beta rays is limited to a small tissue mass, the source of those counted is limited to the portion of the lung contiguous to the plastic window at the end of the tube. This window is of sufficient thickness to remove the I^{131} beta rays. Provision is made for insertion of a 1.6-mm-thick aluminum plate which cuts out all the beta rays but not the gamma rays. If C is the counting rate without the plate and D is the counting rate with the plate then $[C - D]$ is a measure of the P^{32} beta rays from the lung mass.

In vitro measurements of the injected activity and of the radioactivity of blood samples taken at appropriate times are necessary for the interpretation of the *in vivo* measurements. These may be made after completion of the animal experiment. Hematocrit measurements, made as soon as possible after the experiment to insure accuracy, eliminate the necessity of separating cells and plasma for determination of their specific activities.

The principles of *in vitro* measurement for the P^{32} beta rays are the same as those described for the *in vivo* measurement. An 8-ml sample of well-mixed blood is placed in a 30-ml beaker with a standard diameter. The counter is supported over the



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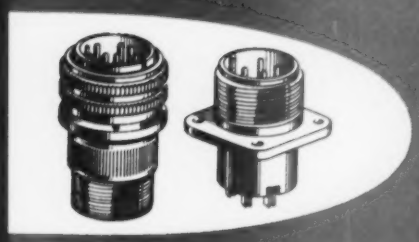

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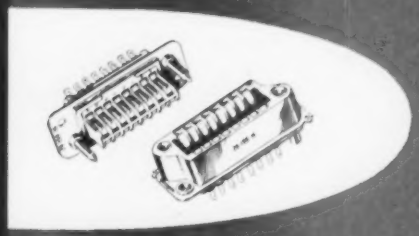

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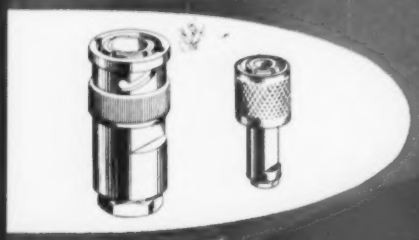

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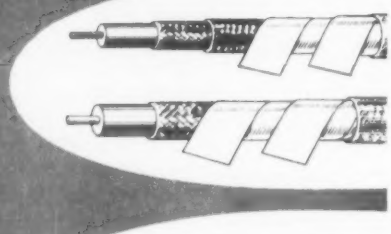

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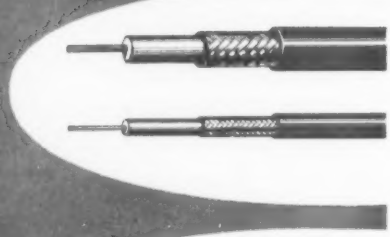

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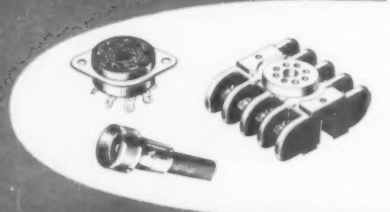

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beaker at a standard height so that there is about 1 cm clearance between it and the top of the beaker. Two measurements are taken: E with 0.56-mm aluminum over the beaker, and F with 2.56-mm aluminum over the beaker. Then $[E - F]$ is a measure of the P^{32} concentration and $[F]$ is a measure of the I^{131} concentration.

Calibration. To relate counts to activities, calibrations are made by measuring standardized solutions of the radioactive iodine and phosphorus. With activity known, conversion factors for *in vitro* and *in vivo* measurements are obtained.

For the *in vivo* calibrations the solutions may be placed in a wide beaker to a depth approximating the effective depth of the radioactivity in a typical dog experiment. All measurements are subjected to the conventional corrections for counter resolving-time, background and radioactive decay where necessary.

Interpretation

In the circulating blood there is a continuous change in concentration of I^{131} in the plasma because of loss of albumin from the circulation, the rapid transcapillary exchanges of intravascular and extravascular water, and loss of I^{131} from the tagged albumin. There are similar changes in P^{32} concentration in the erythrocytes. However, these changes are slow enough so that interpolation of measurements on blood samples taken every 20 min gives reliable intermediate values. From the *in vitro* measurements described previously, if $[F]$ is a measure of I^{131} concentration in a blood sample, then $[F/(1 - H)]$ is a measure of the I^{131} concentration in the plasma, where H represents the hematocrit. Similarly, if $[E - F]$ is a measure of P^{32} concentration in the blood, then $[(E - F)/H]$ is a measure of the P^{32} concentration in the erythrocytes, since P^{32} concentration in the plasma is negligible. From the *in vivo* measurements, if $[A - B]$ measures the gamma rays from a definite lung volume, then $[(A - B)(1 - H)/F]$ measures the amount of plasma in the lung volume. Similarly, if $[C - D]$ measures the beta-rays from a definite lung mass, then $[H(C - D)/(E - F)]$ measures the amount of erythrocytes in the lung mass.

Example. The application of this method is illustrated by the following example: A 15-kg dog was lightly anesthetized with sodium pentobar-

bital and placed in a supine position on a horizontal rack. The counting cylinder was inserted into the pleural cavity against the lung. A suspension of the dog's cells containing 5 mc P^{32} in the erythrocytes and 5 mc I^{131} in iodinated albumin was injected intravenously. After 30 min a sample was taken and a series of counts were made over the lung using the beta- and gamma-ray collimating system. Subsequently *in vitro* measurements were made on the blood sample. The results are given in the table on page 64.

Summary

A method is described for measuring concomitantly *in vivo* the concentration of erythrocytes and plasma in a tissue, a dog's lung being used for illustration. The erythrocytes are labeled with P^{32} and the plasma is labeled with I^{131}-albumin. Determinations are made with a Geiger counter placed in special collimating systems in which differences between counting rates with and without appropriate filters are quantitative measurements of the concentrations.

* * *

This investigation was supported in part by research grant H-1261 (C-2) from the National Heart Institute of the National Institutes of Health, U. S. Public Health Service. The radioisotopes were furnished upon allocation by the U. S. Atomic Energy Commission.

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3. R. C. Gilmore, private communication (1952)
4. A. F. Reid, B. Wilson, NUCLEONICS **10,** No. 10, 72 (1952)

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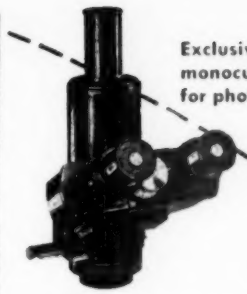

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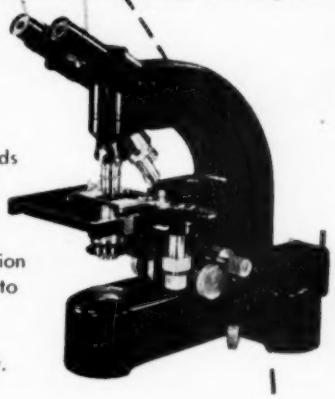

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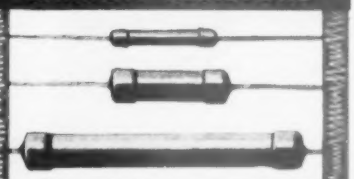

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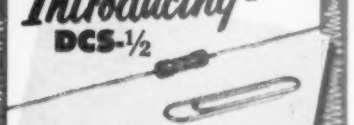

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BOOKS

Basic Methods in Transfer Problems

By V. KOURGANOFF with the collaboration of I. W. BUSBRIDGE (Oxford University Press, London, 1952, xv + 281 pages, $7). Reviewed by H. HURWITZ, JR., Knolls Atomic Power Laboratory, Schenectady, N. Y.

One of the first problems confronting workers in the field of neutron diffusion is that of calculating the neutron distribution near the surface of a scattering medium. This problem is equivalent to the calculation of the radiation density near the surface of a star. The equivalence has proved beneficial to both neutron physicists and astrophysicists, since the neutron physicists have been able to make use of the methods discovered by the astrophysicists and, at the same time, to extend and supplement the techniques along lines which are directly applicable to astrophysics. Kourganoff's book presents, in a unified manner, the methods which have been developed by workers in both fields.

The author's procedure is to illustrate the methods by showing their application to the Milne problem, that is, to diffusion near the plane boundary of a region in which the scattering is isotropic, there is no net absorption, and the mean free path is independent of energy. By dealing with this simplest case, Kourganoff has endeavored to present the fundamental principles of the methods with a minimum of mathematical complexity.

The book consists of three parts. The first part is devoted to establishing the basic concepts and equations. In the second part, the various methods which have been applied to the Milne problem are described. Beginning with the Milne-Eddington approximation, the discussion proceeds to the spherical harmonic method, iterative and variational methods, and finally to methods which yield the exact solution. The section on the Wiener-Hopf method was written by I. W. Busbridge.

In the third part of the book, extensions of the methods to the more general case, the non-gray model stellar atmosphere, are discussed. Although this case is analogous to the energy-dependent neutron diffusion problem, there is little detailed correspondence. The neutron physicist will, therefore, be interested only in the general nature of the methods which are discussed in this section.

Kourganoff has given careful attention to providing detailed explanations and complete discussions of all facets of the problem. This is at once the strength and weakness of the book since the virtues of the scholarly approach are attained with some sacrifice of brevity and conciseness. The casual reader with a background in neutron diffusion may also find it something of an obstacle that the physical arguments are carried through from the standpoint of radiative transfer. Although the relationship between the neutron diffusion quantities and the corresponding quantities in radiative transfer theory are clearly stated, the reader must become familiar with the concepts and symbols of radiative transfer to get the most out of the presentation.

Therefore, the book is not ideal for a person who is looking for a quick review of the theory of neutron diffusion. On the other hand, the student who wishes to delve deeply into the specific problem of diffusion near a boundary will find the book to be of considerable use.

Radioisotopes in Industry

Edited by JOHN R. BRADFORD (Reinhold Publishing Corp., New York, 1953, vii + 309 pages, $8). Reviewed by JOHN L. KURANZ, Technical Division, Nuclear Instrument and Chemical Corp., Chicago, Ill.

This book is a readable, well illustrated presentation of the existing and potential uses of isotopes in industry. In addition, it supplies considerable information about the principles and techniques employed in this relatively undeveloped field. It has two minor weaknesses common to such compilations: lack of uniformity in the level of presentation, and duplication. However, all concerned with the applications of radiosiotopes will undoubtedly find useful ideas and techniques discussed and will obtain some excellent bibliographies for further investigations.

One of the outstanding features of the book are the tables it contains which present in a condensed, well organized manner information which is scattered throughout the literature.

The fundamentals of isotopes are discussed, the fields of application of

isotopes as tracers and radiation sources are well delineated, as are the basic problems involved in specific experiments and the general factors that must be considered in the design of laboratories and the consideration of health hazards.

Some of the discussions and pictures may mislead the uninitiated in connection with the costs, complexity, and hazards involved in the use of isotopes. More emphasis on the type of laboratory used in hundreds of universities and industrial firms rather than AEC installations might have offset this impression.

To the individual starting out in this field, this book should be a useful guide in supplying orientation, basic information, and direction for further study. For those already familiar with this subject, the book contains sufficient new or little publicized information to make it a valuable reference.

BOOKS RECEIVED

Progress in Nuclear Physics, vol. 2, edited by O. R. Frisch, Academic Press Inc., New York (or Pergamon Press, Ltd., London), 1952, viii + 295 pages, $9.25. (*To be reviewed*).

Radiation Chemistry (Discussions of the Faraday Society, No. 12, 1952), Aberdeen University Press Ltd., Aberdeen, England, 1952, vii + 319 pages, 35 shillings. Papers presented and discussion held at Leeds University, April 8–10, 1952. (*To be reviewed.*)

Isotopic Tracers in Biochemistry and Physiology, by Jacob Sacks, McGraw-Hill Book Co., Inc., New York, 1953, vii + 383 pages, $8.50. (*To be reviewed.*)

ALSO OF NOTE

Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water (National Bureau of Standards Handbook 52). Following a discussion of methods of estimating maximum permissible amounts and concentrations and factors that determine the hazards of radioisotopes, this booklet gives tables of maximum permissible amounts for continuous exposure. Most of the remainder of the book is devoted to a table of constants for calculating maximum permissible internal concentration of radioisotopes. There is also a section on equations and nomenclature and a bibliography. *Superintendent of Documents, Washington 25, D. C., $0.20.*

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NEWS

New Industrial Association Formed to Foster Growth of Atomic Energy Industry

This country's first association of industrial organizations working in the atomic energy field was formed last month. Known as Atomic Industrial Forum, Inc., its principal activity will be in the field of information. Formed as the result of a suggestion by T. Keith Glennan, recently resigned member of the Atomic Energy Commission (NU, Dec. '52, p. 83), the forum will seek a broad membership of all businessmen, engineers, scientists, educators, and others involved or interested in atomic energy.

There will be two classes of membership. Regular membership will be open to individuals who are lawfully entitled to access to any "restricted data" as defined in the Atomic Energy Act of 1946. Special membership will be open to individuals, corporations, associations, partnerships, and trusts which are now or expect to be engaged in research, development, or operations in atomic energy. Regular members only will have voting privileges and be eligible to serve on the forum's board of directors.

Glennan, in proposing such an association, declared it "should provide a forum in which industry might develop its best thinking in the interest of the advancement of the peacetime uses of atomic energy; a forum in which industry might develop an informed voice to be heard at governmental levels as new atomic energy policy is hammered out; a forum, not to serve the narrow interests of a few, but to stimulate the industrial development of atomic energy for the good of all."

The organization will have a full-time director and staff. Committees will be built up along technical, engineering, legal, economic, commercial, and management lines.

According to Walker Cisler, president, Detroit Edison Co., the group will in no way act as a lobbying agency. As a project, the forum may make recommendations on changes in the law, but the forum as such will not be a body to endeavor to do that.

As yet the forum has no membership other than those participating in getting it under way. But Cisler says that they have sounded out enough people and organizations to feel sure that there is interest in the organization and that there are many who will become members.

The division of the membership into two groups does not mean, according to Glennan, that those without access to classified information will in any way be "second-class citizens." He feels that the education job to be done by such an organization in assembling and codifying the tremendous amount of existing unclassified information will make the forum worthwhile to non-cleared individuals. Furthermore, Glennan told NUCLEONICS, the forum is not to be a place where members can get classified information that they have not been authorized to have by AEC.

The first meeting of the board of directors is scheduled for this month. At that time incorporation will be complete, and the organization expects to get moving.

Members of the board of directors of the forum are: Albert L. Baker, chairman, Vitro Corp.; Walker L. Cisler; John L. Collyer, president, B. F. Goodrich Co.; T. Keith Glennan, president, Case Institute of Technology; Frederick L. Hovde, president, Purdue University; Alfred Iddles, president, Babcock & Wilcox Co.; Joseph A. Martino, president, National Lead Co.; John R. Menke, president, Nuclear Development Associates; Earle W. Mills, president, Foster-Wheeler Corp.; Mark E. Putnam, vice president, Dow Chemical Co.; Ross W. Thomas, vice president, Phillips Petroleum Co.; Howard G. Vesper, vice president, Standard Oil Co. of California; and J. B. Woodward, Jr., president, Newport News Shipbuilding & Dry Dock Co.

SOME OF DIRECTORS of Atomic Industrial Forum, Inc. Left to right: John R. Menke, Alfred Iddles, T. Keith Glennan, Albert L. Baker, and Earle W. Mills

Public Power Group Asks for AEC Pilot Plant

The atomic power policy committee of the American Public Power Association has called for provision for participation in nuclear power development by publicly owned electric utilities and has urged the Atomic Energy Commission to carry forward development of a pilot plant for nuclear power production.

In a letter to AEC chairman Gordon Dean, the committee also endorsed the provisions of the Atomic Energy Act of 1946 and said that any amendments to the act should provide safeguards against "monopolistic" advantage to private groups. In particular, the committee recommended that:

1. Controls on the use of fissionable materials as stated in the Act, should not be released until adequate knowledge and information are available with respect to the use and development of nuclear energy for industrial purposes and production of electric power. If and when such controls are modified, a provision should be made to prevent any monopolistic advantage

from accruing to any private persons or corporations.

2. AEC should assume responsibility for developing a pilot plant for production of electric energy from nuclear reactors. In view of the large quantity of electric power used by AEC, there is ample justification for expenditure of government funds for design, construction, and operation of such pilot plant or plants, which are needed to advance the feasibility of commercial production of nuclear power.

3. Research in the industrial and electric generation fields and the operation of pilot electric generating plants through several contracts with electrical public utilities and other industrial groups is a suitable means of working out the problems of practicality and economic soundness.

It is urged that publicly owned electrical utilities also be afforded the opportunity of participating in such development contracts because of their experience in distributing such benefits to the public. For similar reasons, the publicly owned electrical utilities should be recognized and be permitted to share adequately in the application of nuclear energy to electric power production when such application has been demonstrated to be appropriate by the results of pilot plant operation.

The letter was sent to Dean by Alex Radin, general manager of APPA, in behalf of the atomic power policy committee, of which Samuel B. Morris, general manager and chief engineer, Los Angeles Department of Water and Power, is chairman.

Submarine Thermal Reactor Reaches Criticality

The Submarine Thermal Reactor, a prototype of the reactor to be used for propelling the submarine Nautilus, has been brought to criticality at the Reactor Testing Station, Arco, Idaho.

Further testing and operation will continue, and the plant will gradually be brought to full power to determine operating characteristics and to train a crew for the Nautilus.

STR and the similar plant for the Nautilus are a joint project of the Argonne National Laboratory, the atomic power division of Westinghouse Electric Corp., and the electric boat division of General Dynamics Corp.

W. STERLING COLE

LEWIS L. STRAUSS

GORDON E. DEAN

. . . These Three Men Will Have Greatest Influence on the Fate of Industrial Nuclear Power

New policy for the nation's atomic energy program is in the making, and the dimensions and timing of the policy shift will be determined largely by three men. The three are Rep. W. Sterling Cole (R., N. Y.), chairman of the Joint Congressional Committee on Atomic Energy; Gordon E. Dean, chairman of the Atomic Energy Commission; and Lewis L. Strauss, special adviser to President Eisenhower on atomic energy and a former member of AEC.

Many other voices will be heard during the forthcoming Congressional hearings on proposed changes in the Atomic Energy Act of 1946. But the decisions of Cole, Dean, and Strauss—and their activities in support of these decisions—will be the most important factors in the fight for new legislation.

Dean almost certainly will be out of the government before all the issues are settled since he plans to retire from AEC in June. But Dean and his fellow commissioners have already drawn up proposals that would permit industry to take a major role in the development of electric generation from nuclear reactors.

Strauss apparently goes along with AEC's ideas on this. He sat in on the Commission sessions at which the proposal was drafted, and quickly won the President's approval of the plan.

This puts Strauss in the position of an advocate of AEC's proposed amendments of the McMahon Act, which would permit AEC to license industrial groups to own fissionable materials and to build reactors for power production. It's just as apparent that Strauss has the full backing of the President—in fact, actually speaks for the White House—on atomic energy matters.

However, the final decision on any changes in the Act lies with Congress. The first step will be the hearings before Cole's committee, and on the surface there would appear to be no snag lurking here. The committee long has been needling AEC to promote further industrial participation in the atomic energy program.

Actually, however, the committee is split almost down the middle on the closely related problem of security. About half the committee members are convinced that no further information regarding developments in atomic energy should be declassified. Just how much declassification will be necessary to make possible any real industrial participation in nuclear power research and construction is debatable. But there surely will have to be some relaxation of classification measures.

Rep. Cole, judging from statements made prior to his election as committee chairman, leans toward relaxation of secrecy where it will help to advance the nation's technology.

Even if he can persuade the rest of the committee to go along with this kind of thinking, however, Cole and his committee have their work cut out for them. The rest of Congress must be persuaded to adopt these views.

Another factor in the Congressional battle will be the appropriations committees. The Joint Committee, time and again, has fought for funds that appropriations bodies, particularly the House committee, have slashed from AEC budgets. This has led to no

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small rivalry between the committees.

This is particularly important now because the present AEC request for funds includes money for construction of one or more experimental power reactors. This request would have been threshed out in advance in the Joint Committee except for the 3-month battle over the chairmanship, which finally was won by Cole.

Now it looks as if the House Appropriations Committee will get word of the proposed AEC experimental power reactors before any other Congressional group. It could deal the cause of industrialization of nuclear power a serious—although not necessarily fatal—blow by turning down any expenditures to promote development of nuclear power.

Congress to Get AEC Plan to Amend Atomic Energy Act; Would Permit Private Ownership of Nuclear Power Plants

The Atomic Energy Commission has prepared recommendations that Congress amend the Atomic Energy Act of 1946 to permit private industry to own fissionable materials and reactor power plants. The plan has the approval of President Eisenhower and the National Security Council and is to be submitted to Congress.

The Joint Congressional Committee on Atomic Energy has been considering for some time the advisability of amending the law so that the private development of nuclear power can proceed and had asked AEC to draw up its recommendations.

Last year the committee held hearings on nuclear power and private enterprise, and more hearings are underway in the present Congress (NU, April '53, p. 66).

The plan, which assumes that the United Nations probably won't adopt international control of atomic energy, calls for two major changes in the law. It would authorize AEC to sell, lease, or loan fissionable materials to private concerns, within a security and safety framework. And it would authorize private concerns to design, construct, and operate reactors to produce power.

If Congress amends the McMahon Act along some such lines, it is expected that the Dow-Detroit Edison group, which has been making a study of the possibilities of reactor power plants, will make a substantial investment in the project. This group favors building a dual-purpose plant for the production of both power and plutonium. The revenue from the sale of the plutonium, which they think should be on a free and open market, would enable them to compete with conventional plants on the cost of electric power.

Council for European Lab Meets in Rome

Nuclear scientists from 12 European nations met last month at the Scientific Research Building in Rome, Italy, for the 5th session of the European Council for Nuclear Research, organized under UNESCO auspices at the end of 1951.

The session, in which 53 representatives took part, continued study of the plans for a European Institute of Nuclear Research, to be built in Geneva, Switzerland, as soon as the final plans are ratified by the governments concerned.

Discussions centered around two main points: technical equipment of the laboratory, and financial and administrative problems.

On the first question, E. Amaldi, secretary-general of the Council, submitted a comprehensive report dealing mainly with the 10- to 30-Bev proton synchrotron to be built for the Geneva laboratory. A 600-Mev cyclotron is also planned for the laboratory. The synchrotron is expected to cost around $11-million, and the cyclotron around $4.25-million.

Construction of the cyclotron is expected to start immediately after the signing of the convention, which should take place at the next session of the council, at Geneva in June.

On the financial problems, it was considered that the amount of $3.9-million a year provisionally set in 1952 for the yearly cost of running the laboratory may not suffice, and the participating governments will be requested to see if the contributions originally set can be increased. It was also decided not to spend more than about $30-million, including administrative costs, until final completion of the laboratory. Building of the laboratory will take 5 to 6 years.

The draft convention among the participating governments, examined at the Brussels meeting in January,

was further discussed, but no agreement was reached on the final text. A six-member working group will meet before the June session to straighten out the remaining difficulties.

Once more, the Swiss delegation obtained assurance that the public will be fully informed on all research at the Center. None of the work at the laboratory is to have any military purpose.

The participating nations were Belgium, Denmark, France, Western Germany, Greece, Italy, Norway, Holland, Sweden, Switzerland, Yugoslavia, and the United Kingdom.

Dean Sees Ample Fuel for Nuclear Power Industry

This country has sufficient fissionable material to support a civilian nuclear power industry and to meet weapons requirements as well. This statement was made by Atomic Energy Commission chairman Gordon Dean in answer to a statement by W. E. Kelley questioning the frequent assumption that with AEC fostering industrial interest in power reactors there will be more than enough uranium available to sustain an industry.

This assumption is usually used to answer the argument that if we are to have a flourishing nuclear power industry we must have enough uranium to fuel the reactors. Kelley, who recently resigned as manager of AEC's New York Operations Office and is very close to the raw materials field, said:

"The recently completed expansion of the atomic energy program, and the expansion now under way, may require all the ore available to us for some time to come, on the basis of national defense. . . .

"Until the conflict between the free world and the iron curtain countries is resolved with suitable safeguards . . . I find it difficult to visualize a situation in which we can afford to divert to power production any significant amount of our basic raw materials, unless there are new and important discoveries . . . While the government may want to give the green light to nuclear power production, I suspect it may be in a position where it can't turn the uranium loose for this purpose."

Commenting to NUCLEONICS on this statement, Dean said: "As I said last September, at a point in the not unforseeable future, we will have, in my opinion, all the weapons we would need to destroy not only the industrial ability of an agressor to make war but

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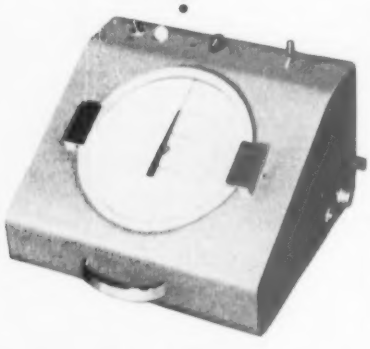

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also his forces in the field. At this point, it is obvious to me that we will have enough fissionable material to support a civilian atomic power industry as well as to meet our own weapons requirements.

"The diversions necessary for this important development are insignificant. And if breeder reactors are developed, which is hoped might be one of the good byproducts of getting industry into the program soon, diversion could never be an issue."

It is expected that a matter as important as this will get as thorough an airing as security permits in the public hearings on industrial nuclear power.

Tripartite Conference Discusses Release of More Reactor Information; Considers Intermediate Classification

Possibilities for releasing more information on reactors as a spur to industrial development constituted the principal business of the 5th tripartite declassification conference held at Chalk River, Canada, on April 8–10. The participants included representatives of the United States, Canada, and the United Kingdom.

Views were exchanged on possible declassification steps that might assist the advancement of nuclear power production. The conference has a strictly advisory capacity, with no power to make commitments for the governments of any of the nations represented.

No other topics covered at the conference have been announced, but it is almost a certainty that the so-called "intermediate band" of classification was one of them. If adopted in the U. S., this plan, already in limited use in Great Britain, would permit the Atomic Energy Commission to release certain information to legitimately interested business concerns without having to resort to clearance of personnel. The companies involved would be held responsible for prevention of publication or other release of such information.

The plan has recently been given serious consideration in AEC and industrial circles. Its appeal is that much data on reactors could be placed in such a classification category at once, with little risk of it's falling into unfriendly hands. Without putting drawings, charts, and other technical data into published form, the information would be of little value to others.

Theoretically, only information with borderline security implications would be put into the new classification. But all information eventually declassified would first pass through this category.

The United Kingdom was represented at the conference by Robert Spence, H. M. Finniston, J. A. Dixon, and R. M. Fishenden. Canada was represented by W. B. Lewis, G. C. Laurence, L. G. Elliott, L. G. Cook, H. H. Clayton, D. G. Hurst, and G. O. Baines.

Representing the United States were the following AEC personnel: J. G. Beckerley, director, office of classification; L. R. Hafstad, director, division of reactor development; Warren C. Johnson, R. H. Crist, T. B. Drew, J. P. Howe, J. M. B. Kellogg, and J. R. Richardson, committee of senior responsible reviewers; C. L. Marshall, office of classification; and E. E. Ferguson.

Plan 3-Day Conference on Nuclear Engineering

A three-day conference on nuclear engineering will be held at the University of California, Berkeley, Sept. 9–11, 1953. The major purpose of the conference is to provide a high-level technical program of contributed papers by and for persons working in nuclear engineering and closely related fields.

Papers will deal with selected, unclassified or declassified results of investigations concerned with the utilization of nuclear reactions. It is expected that the program will feature technical and engineering-economic studies of nuclear power plants, other reactors, isotope production and separation, and isotope applications. There will also be several invited talks providing reviews and integrated discussions of important topics and problems in the field.

Of particular interest for contributed papers are discussions on design, construction, and/or operation of nuclear-process systems; translation of physical, chemical, and biological information into engineering design criteria; and special problems and methods for their solution, including research and de-

velopment techniques of an engineering nature.

The conference is sponsored by the University's departments of engineering at Berkeley and Los Angeles, in cooperation with California sections of the A. I. Ch. E, A. I. E. E., A. S. M. E., and A. S. M., and with the A. S. E. E. committee on atomic energy education.

Persons wishing to present papers should send an unclassified, 200-word abstract, before May 18, to R. A. Fayram, 208 Mechanics Bldg., University of California, Berkeley 4, Calif.

Dunning Sees Economic Nuclear Power in 5 Years

The problems of producing electric power from the fission process are being tackled so well that, according to John R. Dunning, dean of engineering at Columbia University, the principal question that will face builders of power plants 10 years from now will be "is this to be a nuclear or non-nuclear plant?" The answer will be determined, he said, by the availability of conventional fuel.

In a talk last month, Dunning predicted that, within five years, nuclear plants will be quite in line with the costs of more conventional plants. The use of U^{233} (bred from thorium) also holds great promise, he said.

Also, according to Dunning, we are now approaching a period where the entire strength of private industry is needed to proceed with the orderly operation of the atomic energy business. If this is to be done, the Atomic Energy Act must be revised to permit industry participation and provide for private ownership of fuels and production equipment.

Dunning said that a revised Atomic Energy Act must provide a satisfactory climate in which industry will invest more money than at present. He pointed out the existence of widely diverse opinions on how nuclear fuel can be used to produce power, and said that industry must be allowed to experiment with its own money to find the best solution to the problems.

IN BRIEF

• **California calculator.** UNIVAC #5 is being installed at Livermore for use by the University of California Radiation Laboratory and AEC. Purchased with $700,000 of AEC funds, the calculator will be used for both classified and unclassified problems.

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Availability of radioisotopes in large quantities and at reasonable cost has led to their use by many diverse research groups. However, the disposal of radioactive wastes can be a serious economic problem, particularly to small institutions. To relieve isotope users of this responsibility, the four companies listed in the table provide processing and disposal services.

Three of the companies bury waste at sea. Radiological Service ships containers to the Oak Ridge burial grounds.

Crossroads Marine Disposal & Salvage, whose disposal areas are old wreck dumping grounds off Boston and Beverly, Mass., reports that it completely encloses drums and solid waste in 3–6 in. of cement, sufficient to prevent rupture of the container at a depth of 300 ft. Crossroads has found that most unreinforced containers start to crush about 45 ft below the surface. This company is now proposing a plan for large-scale sea disposal of 50-gal drums, or like volumes of solid waste, in a minimum thickness of 12 in. of cement. Dumped beyond the continental shelf, the total weight of 5 tons or more should insure permanent disposal. Crossroads claims that this procedure would be safer and cheaper than incineration.

This new department of NUCLEONICS is devoted to the practical aspects of nuclear technology covering the design, construction, and operation of nuclear reactors and related facilities, and the engineering aspects of the uses of nuclear radiations. Contributions are solicited.

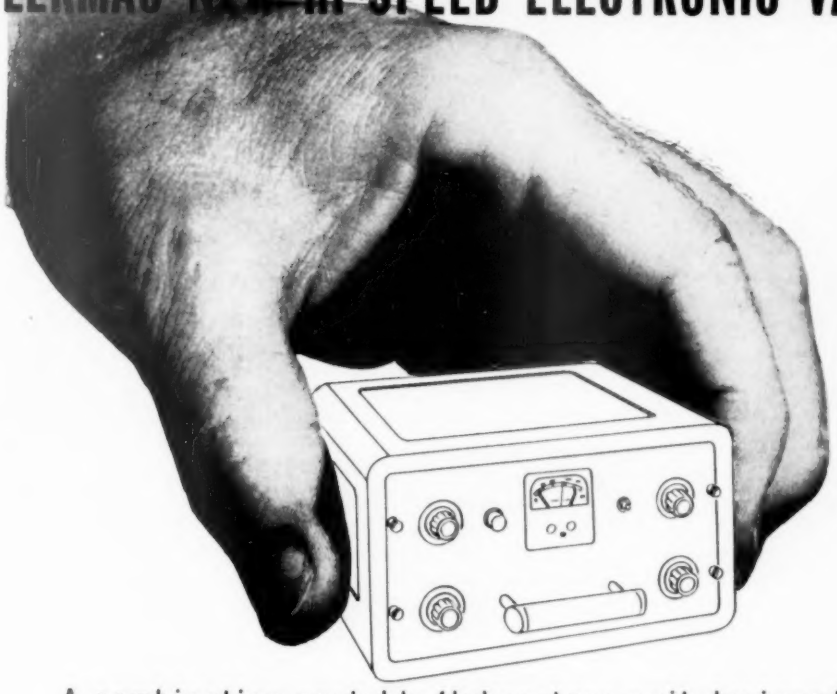
Where a radioisotope user has 400 ft³ of solid waste or more a month, Atomic Energy Waste Disposal Service is interested in installing a process area in the vicinity of the user.

Three of the companies plan to extend their services: Crossroads will serve the U. S. east of the Mississippi; Nuclear Engineering, the state of California; and Radiological Service, the Mid-West. Nuclear now accepts waste from outside of the San Francisco area if containers are furnished and shipment prepaid. All the companies are authorized by the U. S. Atomic Energy Commission to collect waste.

AEC-Authorized Radioactive Waste Disposal Services

Area served	*Activity level handled*	*Volume, physical state, container*
Atomic Energy Waste Disposal Service, 2375 Fruitvale Ave., Oakland, Calif. (affiliate of Nuclear Engineering Co., Inc.)		
All isotope users	Normally up to 3 r/hr; higher under special conditions	Normally 55-gal drums; liquids, solids, contaminated equipment
Crossroads Marine Disposal & Salvage Co., 55 Underwood St., Belmont 76, Mass.		
New England	Specified on container	1–50 gal drums; 12½-gal fiber or steel drum usually suitable for laboratory use
Nuclear Engineering Co., Inc., Box 3159 Rincon Annex, San Francisco 19, Calif.		
San Francisco Bay area	All	Dry waste, 28-gal container; liquid waste, up to 40-gal container (5-gal preferred)
Radiological Service Co., Inc., 92–15 172nd St., Jamaica 33, N. Y.		
Within 100 miles of New York	150 mr/hr at outside of container	5-gal steel containers supplied; all wastes except corrosive liquids

LERMAC NEW HI-SPEED ELECTRONIC VAPOR DETECTOR



A combination portable/laboratory unit designed primarily for the

DETECTION OF MERCURY VAPORS

The Lermac Vapor Detector measures accurately the amount of mercury vapor in the atmosphere. It is engineered to fill the complete requirements of protecting personnel and apparatus against injurious contamination in laboratories, industrial plants and mines. Its sensitivity and accuracy far exceed the normal requirements of safety monitoring.

The Lermac Detector has been developed for maximum efficacy both in the field and in the laboratory and offers a combination of important features not to be found in any other similar instrument.

COMPARE THESE OUTSTANDING FEATURES —

Minimum Drift
High Stability
Accuracy
Light Weight
Cool Operation
Automatic Starting
Continuous Duty Cycle

Equipped with Fresh Air Standard — can be accurately calibrated in a contaminated area.

Audio Alarm — can be preset at any level on both ranges insuring 100 percent protection.

LERMAC INC.

5303 Adeline Street · Oakland, California

SPECIFICATIONS:

Power Required: 115 volts 60 cycles

Power Consumption: 80 watts

Weight: 28 lbs.

Dimensions: 16½ x 9¼ x 10¼ (Excluding Handle & Jumbo feet)
16½ x 10¼ x 15¼ (Including Handle & Jumbo feet)

Balanced Construction: Providing offset carrying handle for maximum convenience when used as portable in field. The position of the handle minimizes cluttering of panel space when instrument is utilized in laboratory.

High Speed Resolution and Recovery: The quick action of the instrument allows varying intensities of vapor to be monitored at the exact concentrations as they occur. Special design features of vapor sampling system give 10 second recovery time when sampling Hg Vapors.

Highly Stable Operation: Obtained by the use of Internal Voltage Regulation, Balanced Bridge Measuring Circuit and Balanced Optical System.

Hg Sensitivity Ranges: Scale A 0-3.5 Milligrams Hg per Cubic Meter
Scale B 0-.1 Milligrams Hg per Cubic Meter

Audio Alarm: This feature allows minimum interference with normal laboratory working procedures. The audio alarm may be set to operate at any level of vapor desired; audio alarm adjustment conveniently located on front panel. Audio Output: 2 watts at 1000 cycles (Loudspeaker).

Zero Drift: When unit has reached its normal operating temperature a drift of less than 1% may be expected regardless of line voltage fluctuations within a range of 105 to 135 volts RMS. The stability of this unit makes it consistent and reliable in operation, and well suited to drive Chart Recorders.

Automatic Start: A special automatic starting circuit is incorporated (for Mercury Light Source) thus providing continuous monitoring regardless of interruption of power.

Conservatively Designed: May be operated for indefinite periods of time.

Want more information? Use post card on last page.

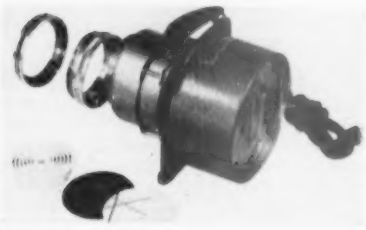
PRODUCTS AND MATERIALS

Direct-Reading Dosimeters

Landsverk Electrometer Co., 550 Garfield Ave., Glendale, Calif. These quartz-fiber dosimeters are provided in six ranges: 200 mr and 5, 10, 20, 50 and 100 r (models L-28, L-41, L-42, L-43, L-44 and L-45, respectively). Metal mounting barrels are ½ in. in diameter and 4½ in. long (200-mr model is 4 in. long). Although the dosimeters are hermetically sealed, they can be disassembled for inspection and repair. All models are charged by the model L-24 charger shown. (P-1)

Deuteroparaffin

Tracerlab, Inc., 130 High St., Boston 10, Mass. This deuteroparaffin, when bombarded with accelerated deuterons, is said to provide a relatively inexpensive neutron source. Available as a white powder, the material melts at 110–120° C and resolidifies to a hard sheet. It has a D/H ratio of 99 and its formula is $CD_3(CD_2)_nCD_3$ where n is approximately 100. (P-2)

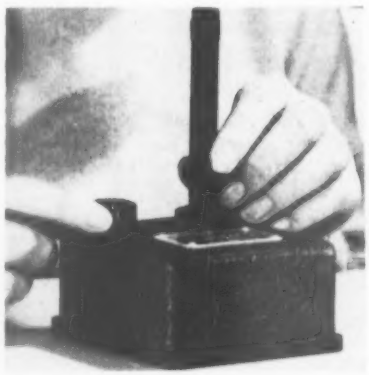


Air Sampler

The Atomic Center for Instruments & Equipment, Inc., 489 Fifth Ave., New York 17, N. Y. The "high volume air sampler" was designed for collection of airborne dust for contamination monitoring. It consists of an interchangeable filter paper system, pump and monitor designed for 24-hr sampling, and a rate-of-flow indicator. Minimum air flow is 30 ft³/min. (P-3)

Alpha-Particle Detector

Radiation Counter Laboratories, Inc., 5122 W. Grove St., Skokie, Ill. The model 2801 detector is designed primarily for monitoring alpha contamination. A panel-mounted loudspeaker produces a "pop" for each alpha particle entering the sensitive volume. Probe sensitive area is ¾ in. by 8¾ in. long. Background is less than 20 cps. High-voltage supply is adjustable from 2,000 to 2,700 volts. Operating voltage is 2,400; plateau length is 200 volts. (P-4)



Dosimeter Charger

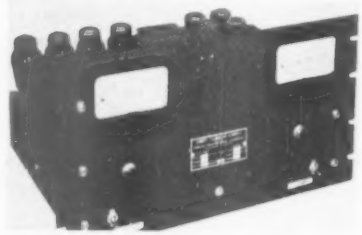
Arnold O. Beckman, Inc., 1020 Mission St., S. Pasadena, Calif. The model 103A charging unit is powered by a single 1¼-volt size D flashlight battery which is claimed to have a life of 6 months to a year under normal use. The charger is designed for any dosimeter having a standard central-charging electrode. A light built into the well allows for reading or resetting the dosimeter. Charging voltage is continuously adjustable between 120 and 240 volts. The entire unit is watertight. (P-5)

Scintillation Counter

Precision Radiation Instruments, 2235 S. La Brea Ave., Los Angeles 16, Calif. The model 112 ratemeter-type scintillation counter is a portable, low-cost counter for medical and laboratory use. Ranges are 0.025, 0.05, 0.25, 1, 5 and 25 mr/hr. Weight of the complete instrument is 15 lb. (P-6)

Shielding Material

P. R. Mallory & Co., Inc., Metals and Ceramics Div., 3029 E. Washington St., Indianapolis 6, Ind. Mallory 1000, a tungsten-nickel-copper alloy, is claimed to be suitable for containers for radioisotope storage, and as a general shielding material for betatron, synchrotron or cyclotron radiation. Pile irradiation tests are said to indicate W^{187} and W^{185} as the primary radiation products. It is also claimed that the alloy has 20% better absorption than lead as a shield for 100-Mev X-rays from a betatron. (P-7)

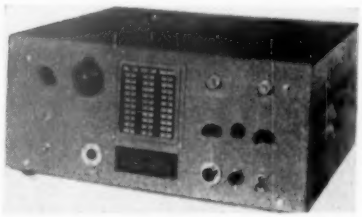


Regulated Power Supply

Kepco Laboratories, Inc., 131-38 Sanford Ave., Flushing 55, N. Y. The model 750 power supply is continuously variable from zero to 600 volts

Want more information? Use post card on last page.

and delivers 0–750 ma, d-c. In the range 30–600 volts, the output voltage variation is $<\frac{1}{2}\%$ for both line fluctuations from 105 to 125 volts and load variation from minimum to maximum current. Ripple is <10 mv, peak to peak. (P-8)



Analog-Digital Converter

Telecomputing Corp., 133 E. Santa Anita Ave., Burbank, Calif. The type 24A analog-to-digital converter has an accuracy of 0.1% (1,000 counts, full scale). The instrument has five continuously variable ranges from 0.02 to 1.0 volts. Digital output can be recorded by punched cards, electric typewriter, magnetic or punched tape. The bridge-balancing circuit requires 0.8 sec or less to reach balance. (P-9)

High-Voltage Supplies

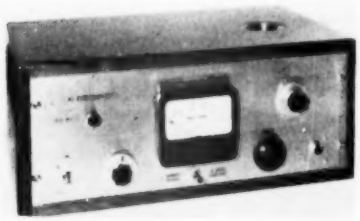
Nucleonics Co. of America, 497 Union St., Brooklyn 31, N. Y. The HV1 and HV2 regulated power supplies provide positive and negative high voltage, respectively, at maximum load of 20 μa. Obtainable outputs are: 900, 967.5 or 1,035 volts, positive, for HV1; same voltages, negative, or continuously variable from 300 to 1,600 volts for HV2. The supplies are powered by two 67.5-volt B batteries and one 1.5-volt A battery. (P-10)

Radiation Indicator

The Victoreen Instrument Co., 5806 Hough Ave., Cleveland 3, Ohio. The model 573 isotope comparator provides a rapid comparison of the relative strength of radioactive materials; activities are indicated as percentage of full scale. Ten aluminum-wall G-M tubes operate from a regulated 900-volt power supply. The instrument is sensitive to beta as well as gamma radiation when a tube shield is removed. With shield in place, sensitivity is said to be maximum for 10 μc I^{131} and minimum for 350 μc I^{131}; six ranges of sensitivity are provided plus zero adjustment. (P-11)

Three-Way Solenoid Valve

Hoppe Engineering Co., Inc., Greensboro, Ind. This three-way electro-hydraulic solenoid, for operating cylinders and power heads, is available in brass, cold rolled steel and various stainless steel alloys. The solenoid coil is claimed to withstand continuous duty at 500° F; the solenoid can be wound for any voltage. The $\frac{3}{4}$-in. valve is 8 in. tall, weighs 6 lb. (P-12)

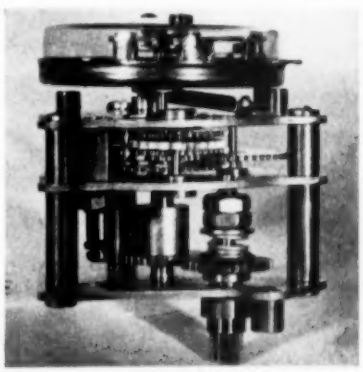


Electrometer

Technical Operations, Inc., 6 Schouler Court, Arlington 74, Mass. The model M-1 electrometer can be used to measure 10^{-14} ampere and is sensitive to 10^{-16} ampere. Accuracy is within ± 1 mv; drift is less than 1 mv/hr. Input resistors are selectable within the range 10^9–10^{14} ohms. An adjustable filtered power supply is provided for ion chambers which can be set in a well on top of the instrument. (P-13)

X-Ray Viewing Window

Bar-Ray Products, Inc., 209 25th St., Brooklyn 32, N. Y. Typical of radiation problems handled by Bar-Ray is this viewing window for protection against X-radiation. It was designed, manufactured and installed by this organization for viewing between the control room and a room containing a 2,000,000-volt X-ray unit. The 24-in.^{2} glass block is cushioned in a structural-steel frame. Lead strips are bonded to the steel frame, connecting the glass limit and the concrete. Outer trim is of stainless steel. (P-14)



Pneumatic Chart Drive

Minneapolis-Honeywell Regulator Co., Industrial Div., Wayne & Windrim Ave., Philadelphia 44, Pa. The Macnick pneumatic chart drive permits operation of the Brown strip-chart recorders in hazardous locations. The all-pneumatic system does not require explosion-proof housings or individually enclosed switch boxes. Chart speed is 1 in. per hr; maximum timing

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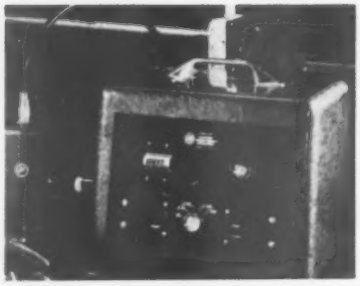
error is ±10 sec/hr. The unit comprises three components: sealed escapement mechanism, gear-reduction unit, and two-way clutch assembly. Air is supplied through tubing connected to rear of the recorder case. (P-15)

Wide-Band Chain Amplifier

Spencer-Kennedy Laboratories, Inc., Dept. NU, 186 Massachusetts Ave., Cambridge 39, Mass. The model 202C wide-band chain amplifier has a bandwidth of 1 kc to 210 Mc. Gain, 20 ± 1.5 db, is uniform over the bandwidth. Risetime is 0.0026 μsec (10–90% amplitude). Input and output impedance is 200 ohms. The noise figure of 9 db allows narrow-band applications. Open-circuit output is 8 volts, rms; phase response is linear within the passband. (P-16)

Portable Leak Tester

Device Engineering Co., 1701 Walnut St., Philadelphia, Pa. These portable manometer leak-test machines, operating on the fall-in-air-pressure principle, are adjustable in the range 0–20 psi within 1 in. H_2O and repeatable within 2 in. H_2O. Regulation is by two pilot-loaded free-diaphragm regulators. A manual bypass valve for filling the tested receptacle is provided; regulator is set to relieve the full bypass valve capacity at a pressure not more than 10 in. above desired delivery pressure. (P-17)



Decade Scaler

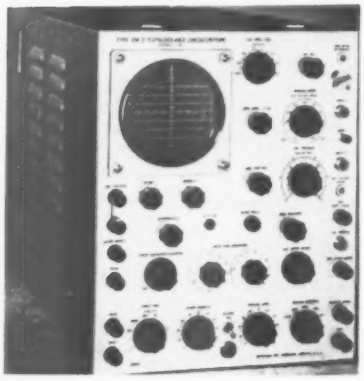
Nuclear Instrument & Chemical Corp., 229 W. Erie St., Chicago 10, Ill. The model 180 low-cost decade scaler incorporates the minimum features necessary for routine counting. High-voltage variation is <0.05% for 1% change in line voltage of 95–130 volts; G-M input sensitivity is ¼ volt. Counts are recorded by a built-in counter; a timer may be plugged into the rear of the unit. (P-18)

Electrical Indicating Meter

Phaostron Co., 151 Pasadena Ave., S. Pasadena, Calif. These hermetically sealed electrical indicating meters were incorrectly described in the March, '53 NUCLEONICS. They operate under conditions of shock, vibration, tumbling and widely diverse weather conditions and are completely interchangeable with all existing "ruggedized" types in a-c and d-c ranges. The meters are manufactured in 1½, 2½, 3½ and 4½ in. sizes. (P-19)

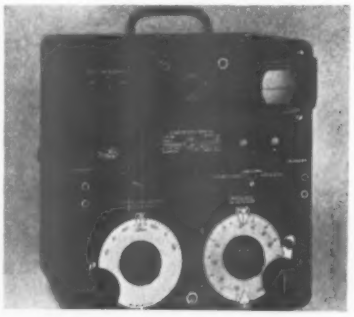
Pulse-Count Converter

Wang Laboratories, 296 Columbus Ave., Boston 16, Mass. The model CC-042 pulse-count converter supplies two outputs of independent conversion ratios, both available at the same time. Each ratio is variable from 0.0001 to 1.0000 in steps of 0.0001, accurate to 1 pulse. Operating frequency is 0–25 kc/sec. Input requirements are 5-volt amplitude with rising time of 10 msec or less; output has an amplitude of 10 volts and duration of 3 μsec. (P-20)



Cathode-Ray Oscilloscope

Tektronix, Inc., Box 831, Portland 7, Ore. The type 514-AD portable oscilloscope provides vertical amplifier bandwidth of 0–10 Mc with 0.3–100 volts/cm sensitivity and bandwidth of 2 cycles to 10 Mc with 0.03–100 volts/cm sensitivity; risetime is 0.04 μsec and signal delay is 0.25 μsec. Time-base range is 0.1 μsec/cm to 0.01 sec/cm within 5% of full scale with 5× sweep magnifier. The unit has direct-coupled unblanking and a flat-faced cathode-ray tube. (P-21)



Comparison Bridge

General Radio Co., 275 Massachusetts Ave., Cambridge 39, Mass. The type 1604-A comparison bridge can be used for testing resistors, capacitors and inductors with an accuracy of 0.1% over the impedance range of 2 ohms to 20 megohms;

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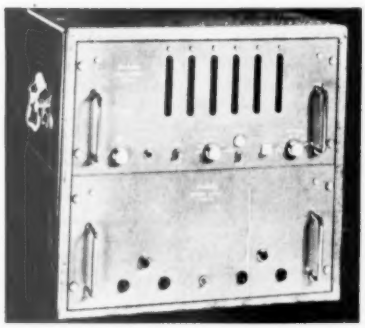
impedance deviation ranges of ±5% and ±20% are provided. The bridge is completely self-contained with a cathode-ray visual detector and an oscillator operating at 1 kc or 5 kc. Operation is from the a-c line. Measurements can be made with the unknown grounded or ungrounded. (P-22)



Electronic Counter

Beckman Instruments, Inc., Berkeley Scientific Div., 2200 Wright Ave., Richmond, Calif. The model 5558 electronic counter counts regularly or randomly occurring events at rates of 20–1,000,000 events/sec with an accuracy of 1 count. The result is read directly on an illuminated bank of six decimal counting units. (P-23)

Regulated Power Supply

Perkin Engineering Corp., 345 Kansas St., El Segundo, Calif. The model M-900 d-c power supply has an output of 300 volts regulated to 0.15% at any current load from zero to 500 ma. Ripple is below 0.004 volts, rms. A time-delay relay withholds d-c output for 40 sec after intial energization to provide for thermal stabilization and for protection of circuits. A d-c bias output of −105 volts is regulated to 1% at loads from zero to 5 ma. (P-24)

Accelerometer

G. M. Giannini & Co., Inc., Bin N, Pasadena 1, Calif. The 24133 ac-

CANNON PLUGS *get good reception*

The high quality audio connectors shown above are available from all Cannon Franchised Distributors. In their great variety of sizes, shapes and contact arrangements there is no problem or technical requirement in the radio, sound, TV or related fields that cannot be met. Cannon plugs are standard on leading makes of audio equipment and microphones.

CANNON ELECTRIC

Since 1915

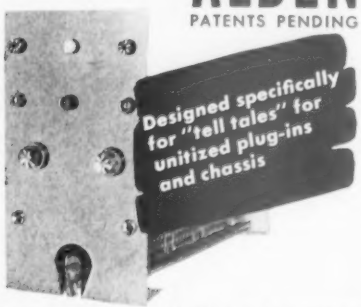
FACTORIES IN LOS ANGELES, TORONTO, NEW HAVEN

Representatives in principal cities. Address inquiries to Cannon Electric Co., Dept. E-124, Los Angeles 31, California.

From Alden's Line of Ready-made Components for Unitized Plug-in Unit Construction:

Miniature Sensing and Indicating Components by ALDEN

PATENTS PENDING



The new trend to unitized plug-in construction calls for tiny sensing and monitoring devices to act as "telltales" when trouble occurs. Device must be rugged, yet take a minimum of panel space. Here are latest in Alden Series to meet this problem.

ALDEN "PAN-i-LITE"

86L

Mounts in .348" hole
Quickly replace bulb from front of panel
Bulb and Lens are 1 pc. instantly replaceable with unbreakable spares.

You can use this new tiny Alden Pan-i-Lite indicator where never before possible. Easily serviced one-piece lens-bulb replaceable from front of panel. Mounts simply by pushing into .348" hole. Gives beautiful indication; glows like a red hot poker. Tiny spares can always be kept ready in kit or taped in recess of equipment. Available in 6V, 12V and 28V, various colors.

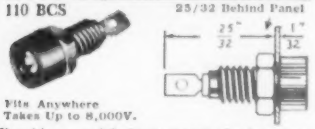
ALDEN "FUSELITE"

440-3/7 FH

Spot trouble instantly. Neon indicator light in lens glows when fuse blows. One-piece molded lens-and-indicator-lite unscrews from front of panel. Ideal for monitoring power to each unitized circuit. Available for 28V, 110V and 250V.

ALDEN TEST POINT JACK

110 BCS



Provides a quick front panel check point for any circuit voltages in your equipment. This tiny Jack fits .257" hole and takes only 25/32" behind panel. Beryllium copper contact withstands hundreds of insertions. Phenolic or nylon insulation, many colors.

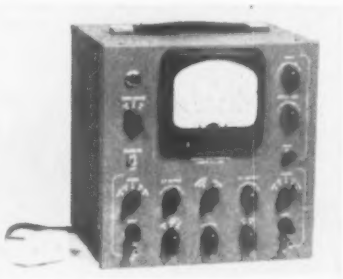
SEND FOR FREE SAMPLES

— also request free "Alden Handbook" 226 pages of techniques and components for Unitized Plug-in Unit Construction.

NA-ALO ALDEN PRODUCTS CO.

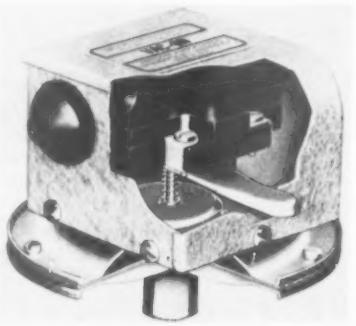
147 N. Main St., Brockton 64, Mass.

celerometer, designed to fit in an oil-filled case, utilizes a potentiometer resistance with a large output, requiring no amplifying unit in most cases. Ranges may be obtained up to 30G; standard resistance ranges are 2,000 or 5,000 ohms. Potentiometer element carries up to 15 ma. Standard instrument has a minimum resolution of 0.25%. It is designed to operate at −54° to 71° C. Typical damping is 0.5 ± 0.075 of critical for a 7.5G instrument. (P-25)



A-C Voltage Analyzer

Advance Electronics Co., Box 394, Passaic, N. J. The type 202 a-c voltage vector analyzer measures the vector sum or difference of two voltages, phase angle between two voltages, and the imaginary and real components of an unknown voltage in terms of a reference voltage. Through the panel binding post, ranges are 8 cps to 2 Mc and 0.06–600 volts, full scale, with 14-$\mu\mu$f impedance shunted by 1 megohm; through the probe, ranges are 2 kc to 500 Mc and 0.6–60 volts, full scale, with 2.5-$\mu\mu$f impedance shunted by 100,000 ohms. Accuracy is ±2% through binding post and ±1 db through probe. (P-26)



Vacuum Switch

Jaycon Associates, 404 N. Washington Ave., Minneapolis, Minn. The Vac-on vacuum switch acts as a circuit breaker or control where the

*Versene**

. . . Decontaminating Agent for Radioactive Deposits

Once anything has been made radioactive, there is no way to destroy this radioactivity. But, radioactive contamination which is usually a deposit of insoluble compounds (metallic oxides) can be solubilized (chelated) with *Versene* and washed away with large volumes of water. This results in great dilution and removes the danger. For deposits on exposed surfaces the addition of a good detergent formulation is recommended. In living organisms *Versene* has been used experimentally to detoxify heavy metals and to help remove radioactive deposits. Send for sample. Write Dept. I for Technical Bulletin No. 2. Chemical counsel available on request.

*Trade Mark Registered

KEY TO CHELATION

"Chemistry's most *precise* chemicals"

BERSWORTH CHEMICAL CO.

FRAMINGHAM, MASSACHUSETTS

Ametron Recording Counter

Consists of two basic electrical circuits, both normally open. Momentary closure of the counting circuit advances indexing device one step. Momentary closure of print circuit causes unit to print accumulated count. Automatic reset to zero available. Wide variety of timing mechanism available as standard equipment.

Unit may be furnished to record elapsed time: 1/10 minute, 1/100 minute, 1 second and 1/10 second. Counting unit may be equipped for direct drive from rotating shaft to count revolutions in units of 1/10. Write for circular SC-24.

THE STANDARD OF ACCURACY SINCE 1908

STREETER-AMET COMPANY

4101 N. RAVENSWOOD AVENUE · CHICAGO 13, ILLINOIS

DEALERS IN MODERN RECORDERS AND SPEED COUNTERS

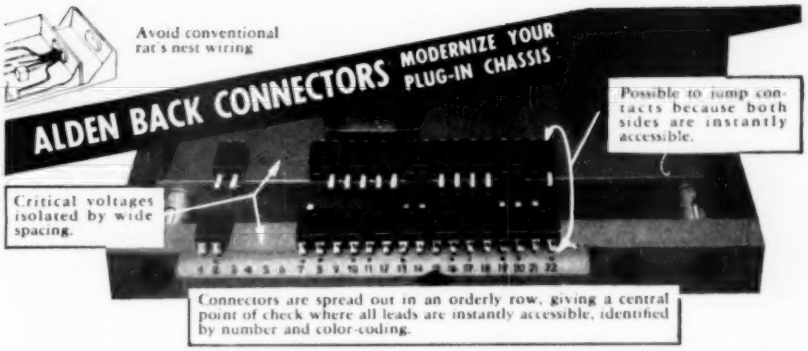
safety of equipment depends on a suction-delivered lubricant, fuel, water or similar liquid; it also serves as a warning control where operating conditions depend on a vacuum. The switch is effective down to pressures of about 1 in. Hg. The model for electric motor use is effective up to 2 hp, a-c, and ½ hp, d-c. (P-27)

Graph Plotter and Follower

Logistics Research, Inc., 141 S. Pacific Ave., Redondo Beach, Calif. The Logrine digital graph follower is a combination input and output device for digital computing equipment, consisting of a digital plotter and a photoelectric graph follower. The follower either follows the curve at maximum rate and delivers electrical impulses corresponding to incremental movements on both axes, or it allows motion on one axis to be externally controlled. Plotter increments are usually $\frac{1}{100}$ or $\frac{1}{64}$ in. Curves down to $\frac{1}{100}$ in. can be followed. Mounted 12 × 18-in. sheets or 12-in.-wide strips can be handled. (P-28)

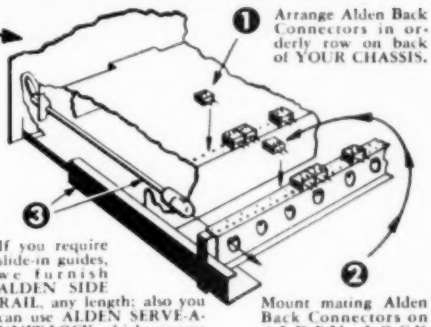
Analog-Digital Converter

Genisco Inc., 2233 Federal Ave., Los Angeles 64, Calif. This analog-digital converter has a standard speed of 250

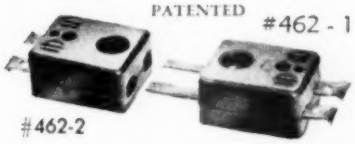

Avoid conventional rat's nest wiring
ALDEN BACK CONNECTORS MODERNIZE YOUR PLUG-IN CHASSIS
Possible to jump contacts because both sides are instantly accessible.
Critical voltages isolated by wide spacing.
Connectors are spread out in an orderly row, giving a central point of check where all leads are instantly accessible, identified by number and color-coding.

It's as easy as this

Modern design no longer tolerates "blind" connectors and congested rats-nest wiring. Use Alden Back Connectors to convert any chassis to plug-in, with 100% circuit accessibility. It's all done with STANDARD components—no design headaches, no waits, no procurement problems.


1
Arrange Alden Back Connectors in orderly row on back of YOUR CHASSIS.
2
Mount mating Alden Back Connectors on ALDEN LOCK FRAME. It's prepunched and tapped to screw in the Back Connectors.
3

ALDEN BACK CONNECTORS



If you require slide-in guides, we furnish ALDEN SIDE RAIL, any length; also you can use ALDEN SERVE-A-UNIT LOCK which engages prepunched Back Frame to pull in and eject chassis.

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- Regulation within 20 PPM* for line voltages of 105 to 130V.
- Load regulation better than 40 PPM* from zero to 500 ma.
- Stability within 100 PPM* per day under average conditions.
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- Short warmup period of 20 minutes.

MODEL 301A. — Voltage: 7.5 to 750 volts. Current: 0 to 500 ma. Ripple: less than 10 millivolts. Auxiliary voltages of —350 and —700 vdc. at 10 ma. —½% regulation, less than 4 millivolts ripple: and 6.3 volts center-tapped at 10 amperes.
MODEL 300N. — Performance same as 301A but voltage range 750 to 3000 vdc. at 0 to 30 ma. No auxiliary outputs.
MODEL 300E. — Performance same as 301A but voltage range —1000 to —1500 vdc. at 0 to 100 ma. Auxiliary output of 6.3 vac. at 1.5 amp.
SPECIAL MODELS. — Special models are available with output voltages from millivolts to kilovolts either positively or negatively grounded and at currents from milliamperes to amperes.

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- High stability — close regulation.
- Electronically regulated.
- High voltage — low current.

MODEL 400B. — Output: 1000 to 5000 vdc. Current: 0 to 1 ma. Regulation against line voltage 105 to 130 v. is within .01%. Regulation against load is .01%. Short term stability is .01% — long term stability is .1%. Ripple less than .01%.
MODEL 400C. — Same as 400B except output: 500 to 1500 vdc.
These models available with positive side grounded.

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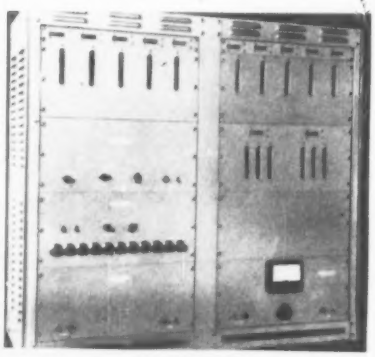
cps; units can be supplied with gear boxes to permit up to 1,600 cps. Output capacity is a function of the number of decades used; means are incorporated for actuating an external relay to permit negative counting. (P-29)

Rotary Vacuum Pumps

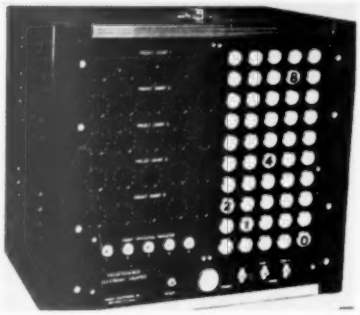
National Research Corp., 70 Memorial Drive, Cambridge 42, Mass. The Rotary Gas Ballast Pump is designed to pump condensible vapors, such as water vapor, without oil contamination or loss of pumping capacity. Gas ballast prevents vapor condensation by bleeding air into the pump and reducing the partial pressure of the vapor, and by heating gases in the pump by compression to raise condensation pressures. Pump capacities are from 2 to 400 ft^3/min; for single units, gas-ballast is effective down to 1 mm Hg. For work below 1 mm Hg, pumps may be used as a combination unit with capacities of 30 to 400 ft^3/min, or as a compound unit (two rotors on the same shaft) with capacities of 2 to 15 ft^3/min. (P-30)



Pulse-Height Analyzer

Radiation Instrument Development Laboratory, 2337 W. 67th St., Chicago 36, Ill. The model 100 pulse-height

analyzer consists of a 5819 photomultiplier and scintillation crystal, a linear amplifier, and a ten-channel discriminator which analyzes pulses between 50 and 100 volts. Channel widths are adjustable to 1, 2 or 5 volts. A built-in pulse generator is provided for calibration. High-voltage supply is regulated to 0.005% for 1% change in line voltage. (P-31)



All-Cold-Cathode Counter

Haledy Electronics Co., 57 William St., New York 5, N. Y. The model GW predetermined decade counter employs an all-cold-cathode-tube design. No warmup period is needed; the Haledy TT-1 tubes used have a minimum life expectancy of 10 yr in operation. Count capacity is 10 digits with a maximum counting rate of 15,000 cpm. Input requirements are 30–70 volts, positive pulse, with maximum rise time of 4 msec and maximum peak duration of 10 μsec. (P-32)

Portable Ultraviolet Lamp

Ultra-Violet Products, Inc., 145 Pasadena Ave., S. Pasadena, Calif. The model B-50-H portable 100-watt ultraviolet lamp operates on 110-volt, a-c, current. Its 3,660-Å beam is said to have a range of 30 ft. The radiation is claimed to be noninjurious. (P-33)

Portable Electric Oven

Grieve-Hendry Co., Inc., 1815 W. Lake St., Chicago 12, Ill. The model PL2A

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Precision Audio Amplifier

This precision audio amplifier, designed and built by Summit Electronics, Inc., of Summit, N. J., is intended for highly accurate laboratory measurements or for high-fidelity home music systems. The unit combines high signal-to-noise and low distortion factors with high power output (distortion less than 1% at full rated 30 watts output). Uses negative feed-back on all stages. With high impedance output, response is flat ± 0.2 db over entire 30 cy to 15 kc range. CHICAGO Sealed-In-Steel Power, Output and Input transformers are specified throughout for the precise and dependable performance required of the instrument.

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S-TYPE

Used in these equipments. Steel base cover fitted with phenolic terminal board. Convenient numbered solder lug terminals. Flange-mounted unit.

Video Distribution Amplifier

The VDA-2 Stabilized Distribution Amplifier is a product of General Communications, Fort Atkinson, Wisc. This rack-mounted video amplifier is designed to accept a video or pulse-type signal at its input and to supply the signal, unchanged in level or other characteristics, to a maximum of four separate channels. The VDA-2 is conservatively designed for maximum stability and continuous operation in TV broadcast service. To achieve absolute dependability, the filament transformer used is by CHICAGO. Where precision and ruggedness are required, you'll find CHICAGO—the world's toughest transformers.

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portable electric oven has damper-controlled temperature to 325° F; air circulation is fan driven. Inside dimensions are 28½ in. wide, 24 in. deep and 20½ in. high. The exterior is contour-formed, heavy-gage steel; insulation is asbestos Aircell. Optional equipment includes outside reading thermometer; 60-min, 6-hr or 12-hr timer; and thermostat control. (P-34)

Pulse-Integration System

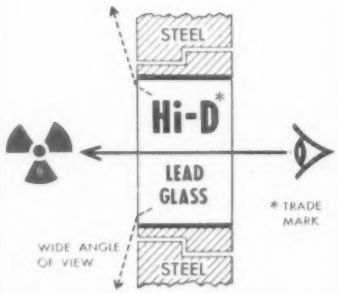
Radioactive Products, Inc., 443 W. Congress St., Detroit 26, Mich. The model E-3 pulse-integration system, designed for fast-neutron studies, consists of a multiple-channel discriminator which injects pulses into a scaling system, weighted according to amplitude. The system is custom made to fit individual requirements. Nondirectional counters can be supplied, as well as remote operation and automatic preset-time counting. (P-35)

Time Totalizer

R. W. Cramer Co., Inc., Centerbrook 5, Conn. The type ET time totalizer is available in two models: one has scale divisions of 0.01 sec over a range of 60 sec; the other has divisions of

Hi-D*

LEAD GLASS WINDOWS IN STEEL WALLS



Hi-D Lead Glass Windows provide clear, direct, wide-angle, binocular vision through heavy gamma-shielding walls. Lead equivalent 0.55; steel, 1.0

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BENCH MODEL 25

• STABLE • DEPENDABLE • MODERATELY PRICED •

- **INPUT:** 105 to 125 VAC. 50-60 cy
- **OUTPUT #1:** 200 to 325 Volts DC at 100 ma regulated
- **OUTPUT #2:** 6.3 Volts AC CT at 3A unregulated
- **RIPPLE OUTPUT:** Less than 10 millivolts rms

WIDTH 14" DEPTH 6" HEIGHT 8" WT: 17 LBS.

For complete information write for Bulletin U-5

LAMBDA ELECTRONICS CORPORATION
CORONA NEW YORK

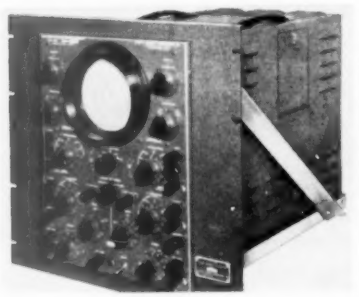
0.01 min over a range of 60 min. The accuracy of each is 0.02%. A differential clutch mechanism avoids the slippage or over-run of a friction-type clutch. (P-36)

Portable Ultraviolet Lamp

Radiac Co., Inc., 489 Fifth Ave., New York 17, N. Y. The M-12 portable ultraviolet light locates fluorescent minerals, such as uranium and mercury, by their characteristic color responses. It is powered by a 6-volt battery and has an operating life of 1,500 hr. A two-way switch and built-in flashlight provide for white or ultraviolet light. The tube is of fused quartz. (P-37)

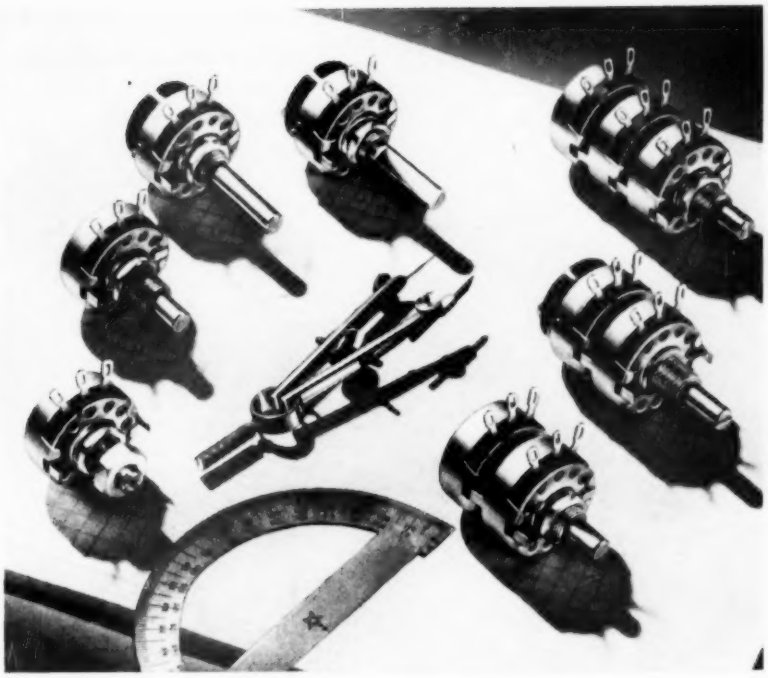
Leak-Test Medium

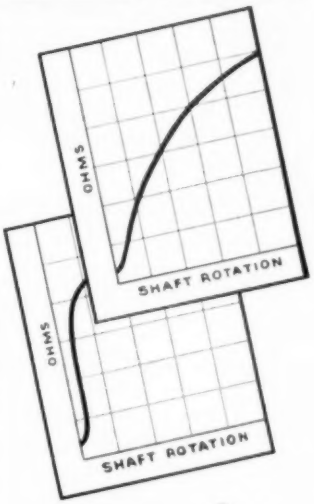
E. I. du Pont de Nemours & Co., Wilmington 98, Del. Freon-12 fluorinated hydrocarbon is a colorless, virtually nontoxic and odorless leak-locating medium. The location of the leak can be found with an electronic device or a gas-burning torch whose flame changes color in contact with nonflamable Freon. Freon provides its own working pressure, about 70 lb/in.^{2} at ordinary temperatures. (P-38)



Oscillograph Adapter

Allen B. Du Mont Laboratories, Inc., Instrument Div., 1500 Main Ave., Clifton, N. J. The type 2598 rack-mounting adapter, for use with Du





Bradleyometer resistor has terminals, faceplate, and bushing molded in the plastic body.

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Unaffected by Heat, Cold, Moisture, or Long Use

For circuits requiring a top quality adjustable resistor not affected by moisture, heat, cold, or age . . . the Allen-Bradley Type J Bradleyometer is the ideal unit.

The resistor element is molded as a single piece. It is not a film or paint type of resistor. Because of its nature, the resistor can be built up to satisfy any resistance-rotation curve. After molding, the resistor is no longer affected by heat, cold, moisture, or age. There are no rivets . . . no welded or soldered connections . . . and the shaft, cover, faceplate, and other ferrous parts are made of corrosion-resistant metal. Let us send you the latest data on Bradleyometers.

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A-B Quality

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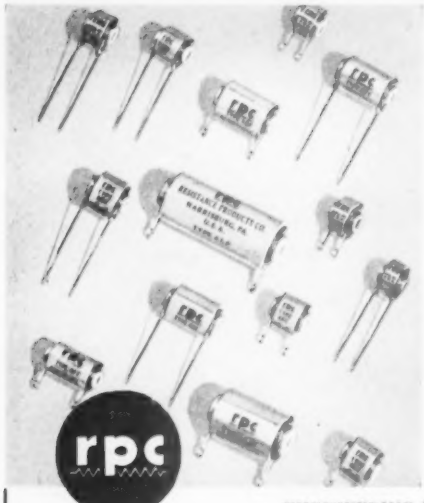

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RPC Precision Resistors meet Government specifications. They are acceptable for all types of equipment—test instruments, electronic computers and scientific equipment. Requirements of Jan-R-93, Mil-R-93A are fully met. Advanced methods of production have made possible large or small orders at reasonable cost with prompt delivery. Write for catalog and helpful information about RPC's resistors.

ENGINEERING DATA

RPC TYPE	Government Specification		DIMENSIONS (inches)		RESISTANCE OHMS			WATTS	
	JAN-R-93	MIL-R-93A	LENGTH	DIAMETER	Min.	MAX. WITH LOW T.C. ALLOY .0015 Dia.	.001 Dia.	JAN or MIL	COMM'L
AFB*	RB10	RB15	17/32	17/32	0.1	160 MEG	650 MEG	.25	.5
AGB*	RB10	RB15	17/32	5/8	0.1	235	1.0	.25	.5
AFC*	RB11	RB16	5/8	17/32	0.1	225	1.0	.33	.5
AGC*	RB11	RB16	5/8	5/8	0.1	330	1.5	.33	.5
AFF*	RB12	RB17	1	17/32	0.1	475	2.0	.5	1
AGF*	RB12	RB17	1	5/8	0.1	700	3.0	.5	1
AJS	RB13	RB18	17/32	13/16	0.1	1.25	5.0	.5	1
AJP	RB14	RB19	21/16	13/16	0.1	2.5	10.0	1	2

*NOTE Can be furnished with 1½" long 20 guage tinned wire leads instead of lug terminals. Suffix "W" after type denotes wire leads.

RESISTANCE PRODUCTS CO.

714 RACE STREET **HARRISBURG, PENNA.**

Precision Wire Wound–High Voltage–High Megohm–High Frequency–Hermetically Sealed–QUALITY RESISTORS

encapsulated

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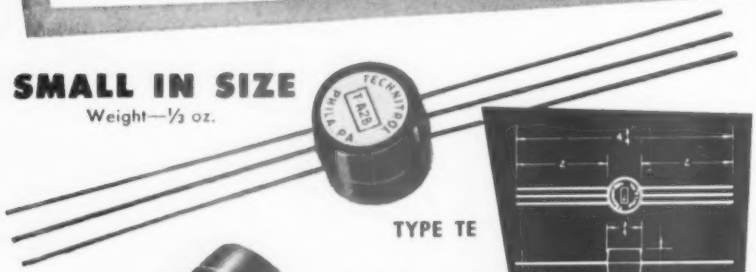
wound to your requirements

SMALL IN SIZE

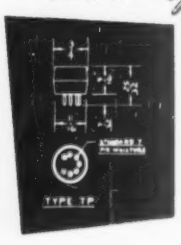
Weight—⅓ oz.



TYPE TE

TYPE TP

Write for Bulletin 166N for additional information and specifications.



SMALL IN PRICE

As low as $5.00 each in quantity lots. One to three—$7.50 each.

TECHNITROL

ENGINEERING COMPANY

2751 N. 4th St., Philadelphia 33, Pa.

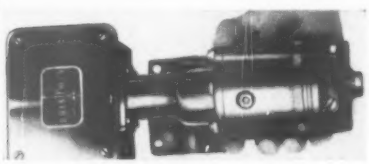
Mont types 303, 303-A, 303-AH and 322 cathode-ray oscillographs, bolts to a standard 19-in. relay rack. Channels fastened to the adapter provide support for the instrument; provision has been made for bolting the instrument cabinet to the channels so that the entire relay rack may be moved with the instrument in place. (P-39)

High-Speed Camera

Charles A. Hulcher Co., Inc., 40 Manteo Ave., Hampton, Va. This camera takes pictures on 70-mm film at up to 25 5-in. frames/sec or 50 2½-in. frames/sec. The camera is equipped with reflex focusing, variable frame-speed control, electric braking and changeable frame size. The unit weighs 32 lb and uses perforated film in 100-ft daylight-loading spools. (P-40)

Potentiometers

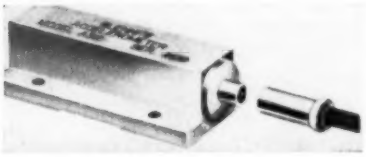
Helipot Corp., S. Pasadena, Calif. The model L continuous-rotation potentiometers include the model L with bushing mounting and sleeve bearings, model LS with servo lid mounting and Oilite bearings, and model LSP with servo lid mounting and ball bearings. Models L and LS have linearities of ±0.5% in ranges from 10k to 100k ohms; linearities to ±0.1% (5k ohms and above) are made to order. Phasing is done with an external ring clamp; sections of ganged assemblies can be phased independently. Up to 33 taps can be made in a section. (P-41)



Pneumatic Transmitter

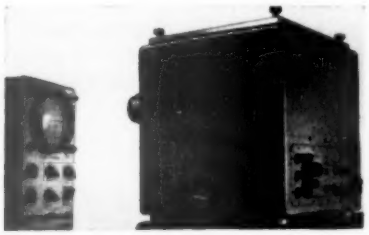
The Bristol Co., Sales Promotion Dept., Waterbury 20, Conn. The series 650 pneumatic transmitter measures and transmits readings of temperature, pressure, vacuum, differential pressure, and liquid level to recording, indicat-

ing, and controlling receivers. Transmission is by means of 3–15 psi air pressures. The transmitter is sensitive to as little as 0.03% of range, including reversal. (P-42)



Accelerometers

Gulton Mfg. Corp., Metuchen, N. J. The A310, A311 and A312 accelerometers are usable over an acceleration range of 0.01–500 g. They can be used up to frequencies of 2,500, 1,200 and 800 cps, respectively. The three units are mechanically interchangeable, using the same mounting holes. Within the temperature range −40° to 80° C, sensitivity varies $< \pm 10\%$. (P-43)



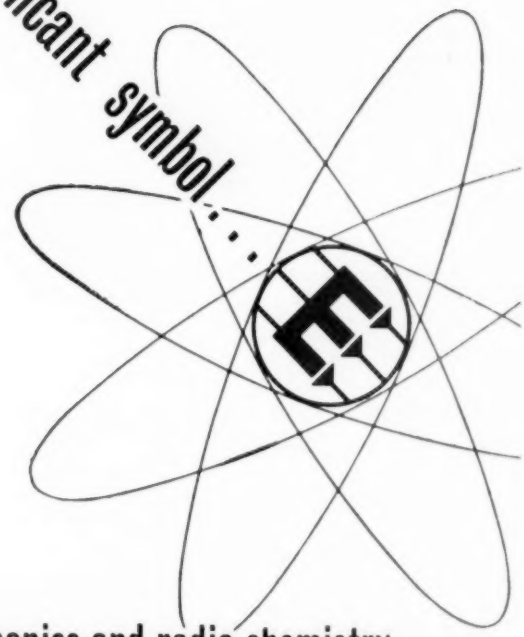
Oscilloscope Recording Camera

J. A. Maurer, Inc., Photographic Instrumentation Div., 37-01 31st St., Long Island City 1, N. Y. The M731 cathode-ray oscilloscope recording camera provides for still, moving film and drum photography. The unit accommodates 35-mm or 70-mm film or paper. The drum attachment carries 20 in. of film at 4–1,200 in./sec; the continuous-feed mechanism handles 25 ft of film at 0.4–100 in./sec. (P-44)

Thermal Radiometers

Beckman & Whitley, Inc., 974 San Carlos Ave., San Carlos, Calif. The model 188 radiometer performs total-hemispherical or net-exchange radiant-heat measurements. The thermopile heatflow meter is 4½ in. square; con-

The most significant symbol . . .



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***Manufacturers:*—EKCO ELECTRONICS LTD, SOUTHEND-ON-SEA, ESSEX, ENGLAND**

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Model 100

Square Pulse Generator

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For Nuclear Pulse Work, Radar, TV, Wide Band Amplifiers, and in the design, calibration and servicing of fast electronic systems:

NOW—A new Square Pulse Generator with a rise time of one millimicrosecond (10^{-9} seconds) and a pulse width which can be varied from one millimicrosecond to several microseconds provides the ideal test instrument for fast electronic circuits. Both positive and negative pulses of a 100 volts maximum amplitude, into low impedance (such as 50 ohm cable) are generated; the pulse amplitude can be varied from 100 volts to .006 volts in 1 decibel steps by means of selector switches on the front panel. One, two, or more pulse outputs, each of which can be individually attenuated and delayed are available in various models.

For further details, write for bulletin P-24 or contact our Engineering Division.

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vection effects at the heatflow transducer are eliminated by an air blower. Transducer output is sufficient for the direct driving of a recording galvanometer. As a total-hemispherical radiometer (model 188-1), a double shield is provided below the transducer; for net-exchange measurements (model 188-2), upper and lower surfaces of the transducer are blackened. Instrument reaches 95% of steady-state conditions in 12 sec. (P-45)

INDUSTRY NOTES

▶**Westinghouse Electric Corp.,** 401 Liberty Ave., Pittsburgh 30, Pa., has appointed Edmund T. Morris assistant manager of its atomic power division. Mr. Morris was formerly manager of subcontracting in the Westinghouse defense products organization.

▶**Isotope Products Ltd.,** Oakville, Ontario, Can., has formed Isotope Products Inc. in association with Petrol Structors, Inc., of New York, Houston, and Tulsa. The new company is initially offering radiographic inspection services to the U. S. petroleum industry. Its headquarters are in Houston, Texas.

▶**Radioactive Products, Inc.,** 443 W. Congress St., Detroit 26, Mich., completed its first year of profitable operations in '52. Net sales were $287,437 (93% over '51); gross income was $59,235, and net income was $1,586.

▶**Leeds & Northrup Co.,** 22nd and Locust Streets, Philadelphia 3, Pa., together with S. A. Integra, Liége, Belgium, has formed Integra-Leeds & Northrup Ltd., Birmingham, Eng. The new firm will be agent and manufacturer of L&N and S. A. Integra products.

▶**Perkin Engineering Corp.,** 345 Kansas St., El Segundo, Calif., has moved into a new 10,000 sq. ft. plant.

▶**Nuclear Instrument & Chemical Corp.,** 229 W. Erie St., Chicago 10, Ill., has appointed George A. Fink and Murray E. Volk as technical associates in research and chemistry, respectively.

▶**Freed Radio Corp.,** 200 Hudson St., New York 13, N. Y., has changed its name to Freed Electronics and Controls Corp. The "Freed-Eisemann" trademark will be used on nucleonic and electronic equipment.

LITERATURE AVAILABLE

Flow-test kit. Catalog 33 describes a

portable, wide-range kit. *Fischer & Porter Co., Hatboro, Pa.* (L-1)

Dehumidifying equipment. Folder enables determination of equipment and capacity needed. *Remington Corp., Auburn, N. Y.* (L-2)

Radiation measuring instruments. Catalog G-6 describes roentgenmeters, dosimeters and other devices. *Landsverk Electrometer Co., 550 W. Garfield Ave., Glendale 4, Calif.* (L-3)

Lubricants and sealants. Bulletin 30 describes Fluorolubes (chlorotrifluoroethylene polymers) for uses requiring lubricity and high density. *Hooker Electrochemical Co., Niagara Falls, N. Y.* (L-4)

V-belt pump unit. Catalog 107 gives details on unit for handling noncorrosive liquids. *Tuthill Pump Co., 939 E. 95th St., Chicago 19, Ill.* (L-5)

Pulse transformer. Sheet T-36 gives specifications of transformer for use in microsecond range. *Berkshire Laboratories, 538 Beaver Pond Rd., Lincoln, Mass.* (L-6)

Wire filters. Bulletin 531 illustrates uses of wire cloth. *Multi-Metal Wire Cloth Co., Inc., 1350 Garrison Ave., New York 59, N. Y.* (L-7)

Air-operated controllers. Bulletin A120 describes instruments for automatic control of pressure, temperature, liquid level, flow, humidity, *p*H, and electrical variables. *The Bristol Co., Waterbury 20, Conn.* (L-8)

Circuit assembly. Manual C3 gives information on wiring and assembly of electrical connectors and components. *American Phenolic Corp., 1830 S. 54th Ave., Chicago 50, Ill.* (L-9)

Thermal time delay relays. Bulletin 30 describes miniature and octal-size hermetically sealed adjustable relays. *G-V Controls, Inc., 28 Hollywood Plaza, East Orange, N. J.* (L-10)

Mass spectra data. Punch cards with data for 550 compounds described in bulletin CEC 1823. *Consolidated Engineering Corp., 300 N. Sierra Madre Villa, Pasadena 8, Calif.* (L-11)

Air cleaners. Catalog E-60 describes operation of and gives data on electronic air cleaners. *Trion, Inc., 1000 Island Ave., McKees Rocks, Pa.* (L-12)

Microwave equipment. Catalog C2 explains microwave concepts and lists equipment. *De Mornay-Bonardi, Inc., 3223 Burton Ave., Burbank, Cal.* (L-13)

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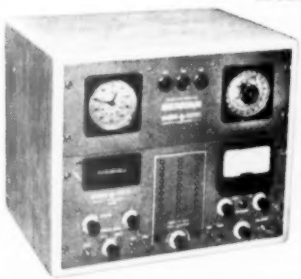

The MULTISCALER

Model 1070A (shown) is the newest model of the *first complete automatic scaler*. Available with either a scale of 1000 or 4096 the Model 1070A also has a choice of 5 microsecond or 1 microsecond resolution. Other features include: preset time, predetermined count, electric register, scale selector, built-in pulse amplifier, discriminator, regulated 500 to 3000 volt power supply, provision for remote operation and many others. Atomic offers a choice of complete binary or decade scalers to meet every laboratory requirement.

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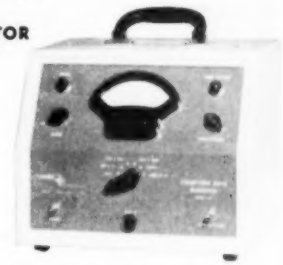

Model 410 (shown) for rate metering in laboratory radioactivity work has an accuracy of 2%. It is equipped with a standard Geiger probe input (0.25 volt sensitivity). The Model 410 also has an input connection for preamplifier or scintillation probe which provides 300 volts to plate and 6.3 volts to filament, and high voltage. The inclusion of a discriminator permits sine wave, square wave and pulse counting without affecting calibration.

Controls include: range switch 0-200, 0-2,000, 0-20,000 – 200,000 CPM; high voltage supply, 400 to 2000 v @ 1 ma., 0.05 per cent regulation; Calibrate switch (7200CPM).

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Nuclear-Chicago is the new *trade* name adopted by Nuclear Instrument and Chemical Corporation of Chicago to positively identify its products and services, and to eliminate confusion in the minds of our friends.

Remember Nuclear-Chicago—not just Nuclear—when you think of precision instruments for radioactivity measurements, and helpful services like the new Nuclibadge personnel protection system.

NEW FILM BADGE SERVICE

Coincidentally with the new trade name, Nuclear-Chicago is also announcing its new film badge service for personnel protection. This service supplements the pocket chambers now available to provide a complete service for every radioisotope user.

Two types of service are available on a subscription basis:

1. The "Nuclibadge" system is Nuclear-Chicago's complete film badge service for accurate interpretation of gamma and X-ray dosages. The exclusive Nuclibadge has an open window for estimating beta exposures, with a special design feature to minimize the back scatter error found in other badges.

Lightweight plastic badges are furnished in three colors on a rotating weekly basis. Nuclear-Chicago takes care of processing, interpreting, and reporting dosages quickly. On request, excessive exposures are reported immediately by telegram.

2. For those already using Oak Ridge type badges, Nuclear-Chicago offers a special service to supply identified film packets each week and handle processing, interpreting, and reporting on these film packets.

COMPLETE PROTECTION

Under both types of service, positive identification of the wearer is provided on both the outside of the film packet and on the film emulsion. With three or more badges, one blank control badge or packet is provided without extra charge.

In addition, only Nuclear-Chicago guarantees measurement of accidental exposure to large amounts of activity. Each film packet contains an exclusive high range "catastrophe" film which is processed only in case of high exposure of lower range films.

An additional Nuclear-Chicago feature is the availability of any number of badges. There is no minimum quantity required.

AUXILIARY SERVICE

Part of the standard procedure for film interpretation is the processing of five calibrated films with each group of films to assure accurate interpretation of dosages. When requested, and at additional cost, an extra set of calibrated films will be processed with the films, used for interpretation, and shipped to the subscriber when the films are returned.

Complete information on the new film badge service will gladly be sent on request. Address inquiries to **Film Badge Service, Nuclear Instrument and Chemical Corporation, 235 West Erie Street, Chicago 10, Illinois.**

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